Materials for Gas Cooled Reactors

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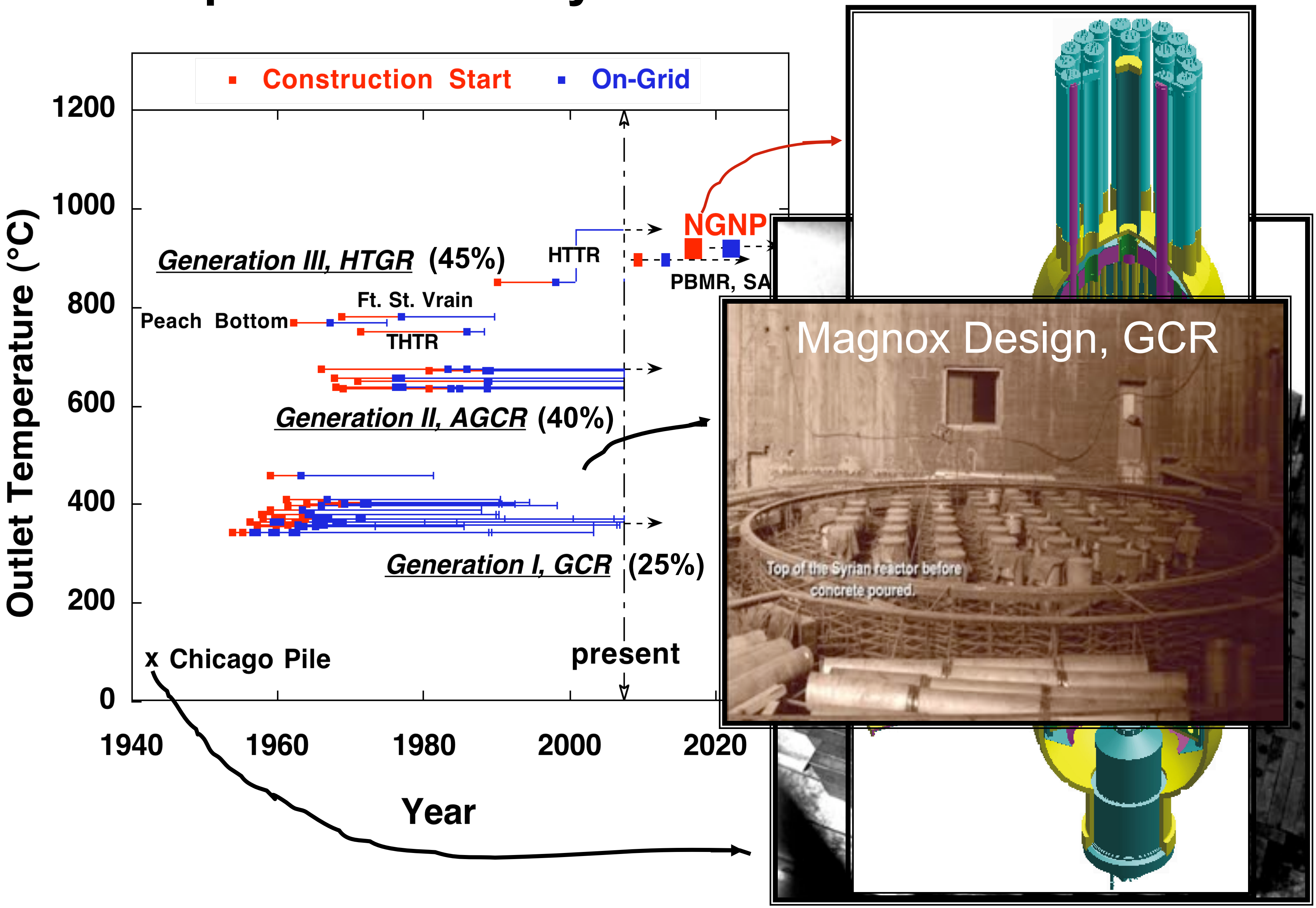
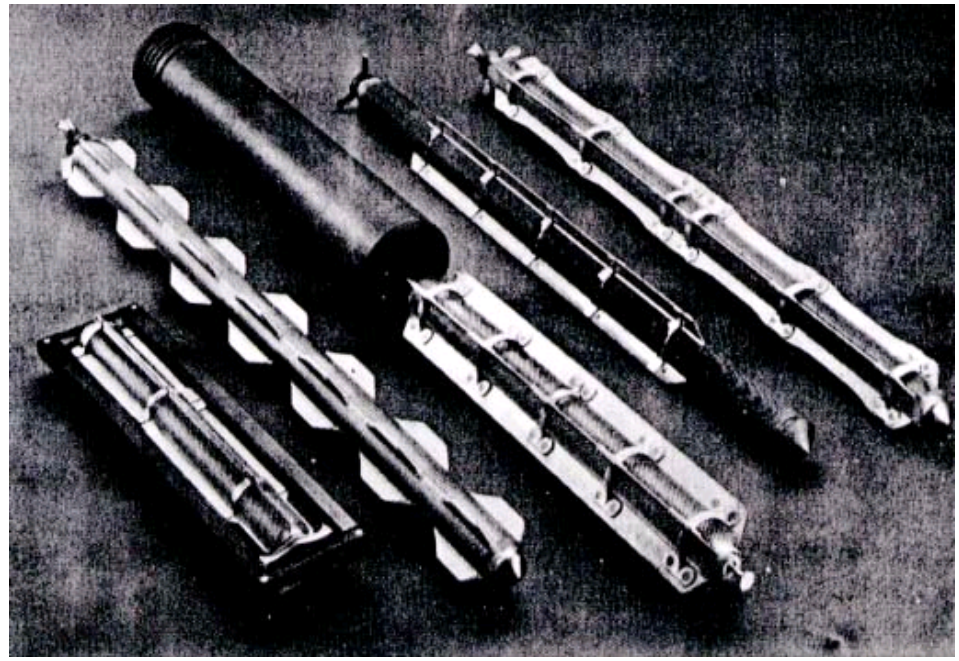
Jeremy Busby, Roger Stoller and Yuri Osetskii (ORNL)
Kobus Smit (PBMR, Pty Ltd.)

Temperature History of Gas-Cooled Reactors



MAGNOX Design (~400°C)

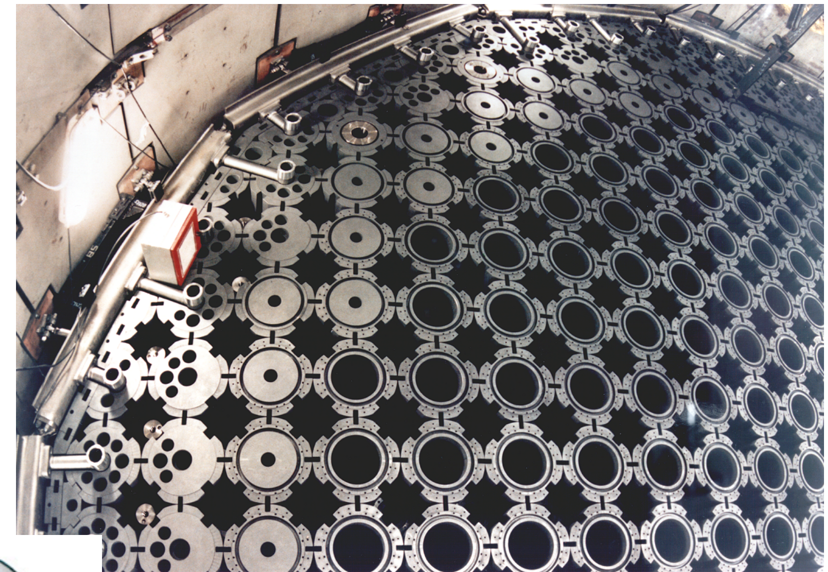
- **MAGNOX are Gas cooled/ Graphite moderated reactors with natural uranium fuel which is not chemically compatible with CO2 coolant.**
- **Rapid oxidation would occur if the temperature exceeded about 250°C. Therefore it is necessary to clad the bar, protecting the fuel from oxidation and avoiding contamination of the coolant.**
- **A non-oxidising alloy of magnesium hence the acronym Magnox (MAgnesium Non-OXidising) was chosen because of its properties**
  - **Compatible with Uranium**
  - **Compatible with CO2**
  - **Very low neutron capture cross section**
  - **Good thermal conductivity (heat transfer)**
  - **Good Strength**
- **The magnox fuel element does not normally operate at temperatures in excess of 450°C, approximately 200°C below the melting point (640°C).**
- **Concrete and Steel Containment/Pressure Vessel**

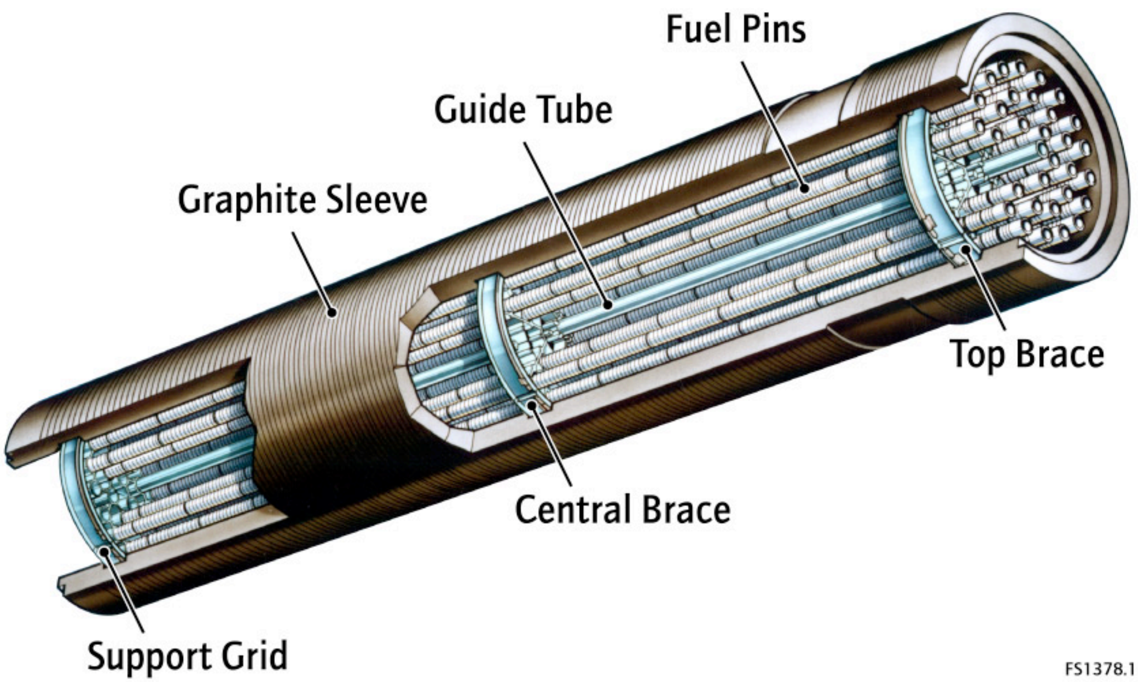


Examples of Magnox Fuel Cladding Design

AGR Reactors (~ 600°C)

- Higher Enriched (4%) UO2 fuel
- Stainless Steel fuel cladding
- Graphite moderated, CO_2 coolant
- Indirect steam cycle electricity
- Boronated Stainless Steel control rod
- Concrete and steel containment/ pressure vessel

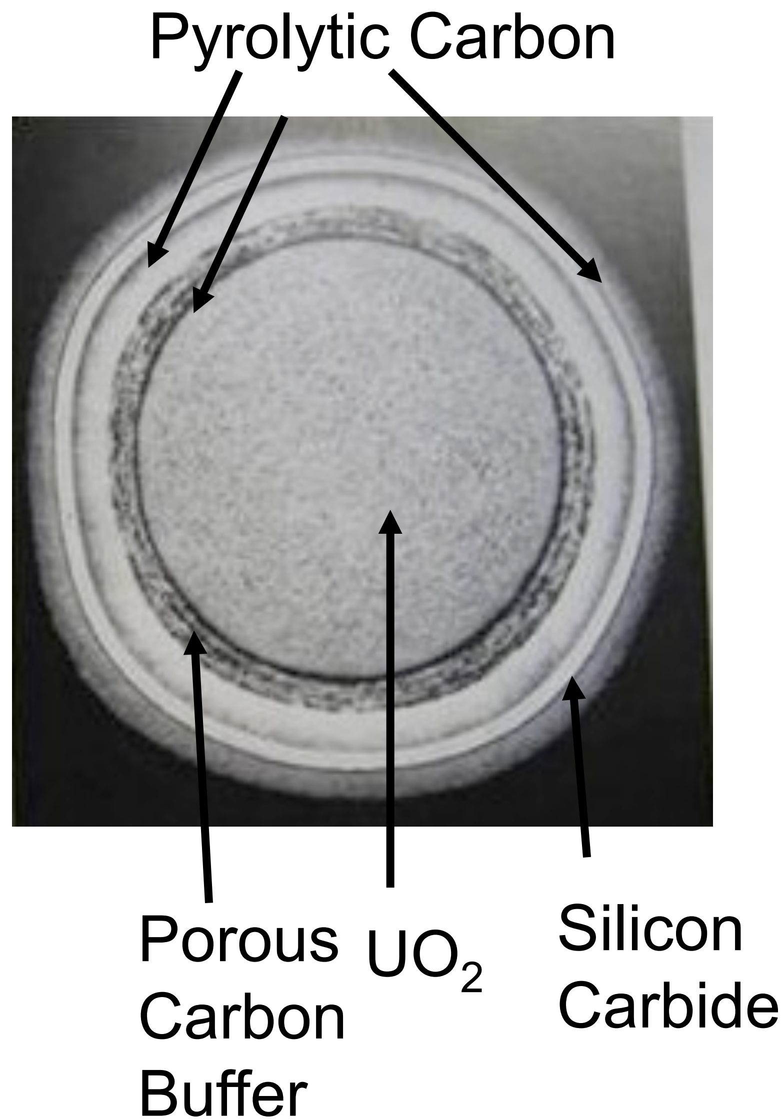




Higher Effeciency
~ 40%

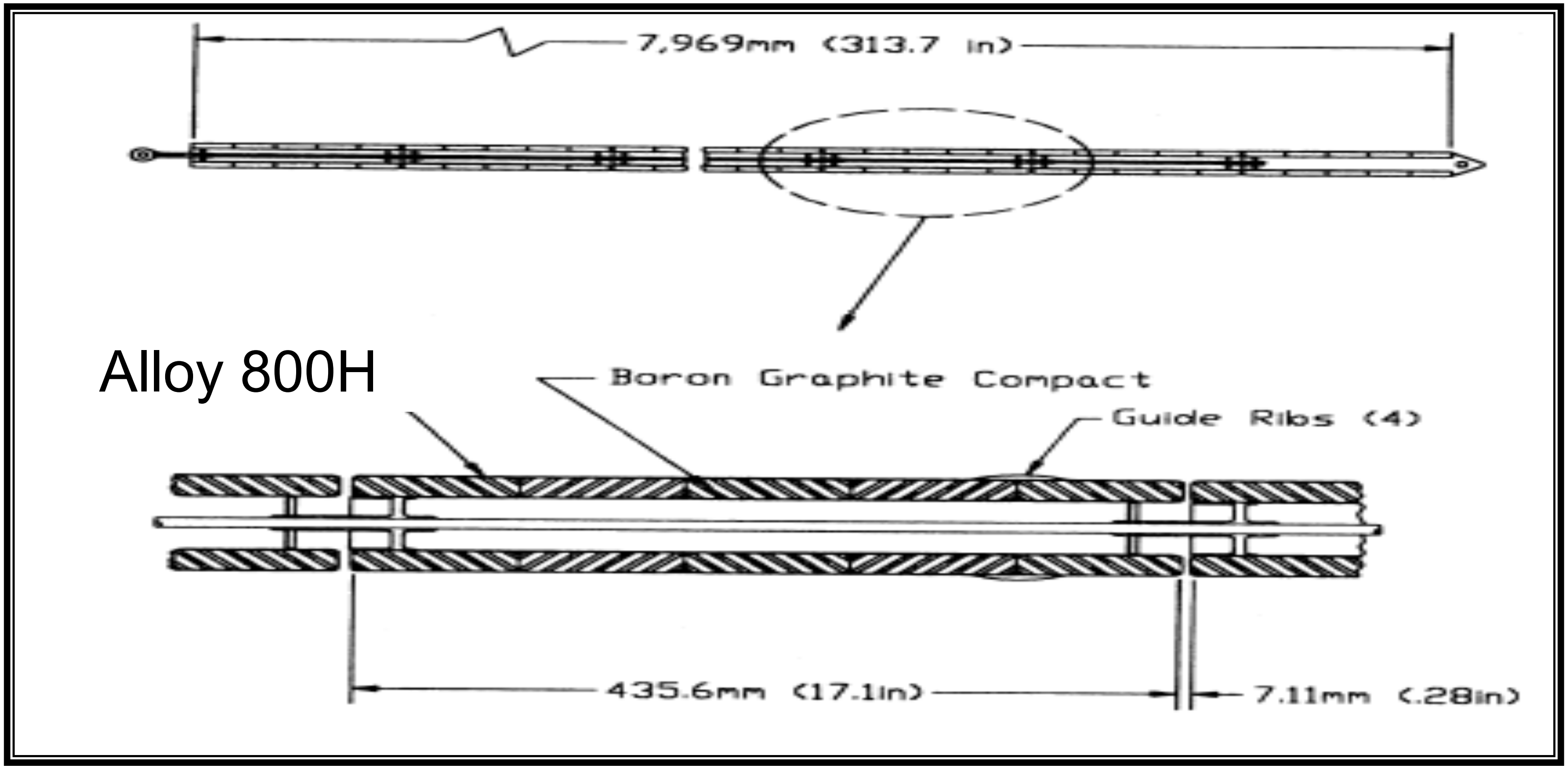
HTGR Clad

- **HTRs –have an outlet temperature in excess of 750 deg C and are graphite moderated/ helium cooled**
- **Helium is to all intents and purposes inert both chemically and neutronically.**
-
- **The high temperatures have led to an all ceramic fuel particle with a silicon carbide layer providing the cladding.**
- **SiC has a very high operating temp. Each fuel particle has it's own barrier to fission product release – under upset conditions integrity is guaranteed up to 1600 ^{o}C.**
- **A typical TRISO particle is shown (<u>Tr</u>iple-coated UO_2 particle: <u>I</u>nner Pyrocarbon – <u>S</u>iC – <u>O</u>uter Pyrocarbon)**
- **Each particle is sub 1 mm in diameter**



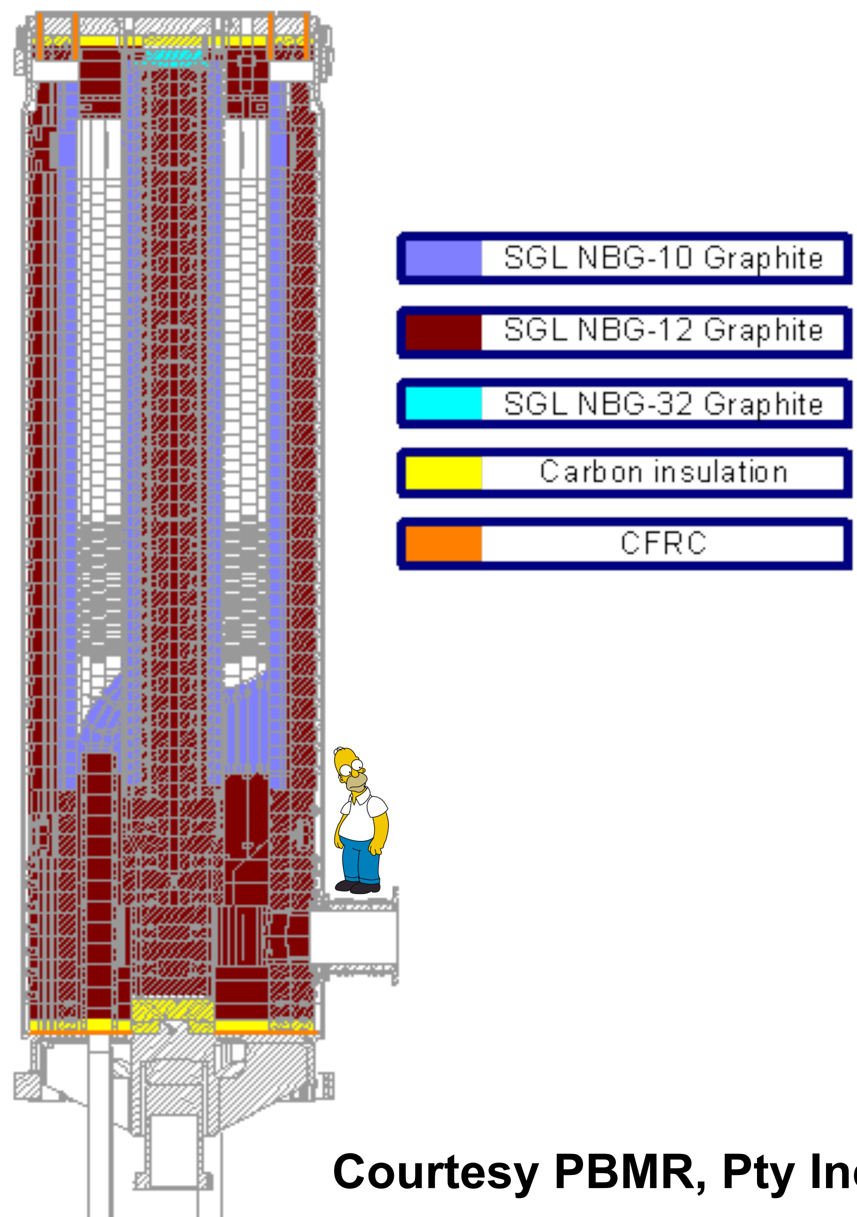
HTGR Control Rod Concept

(Courtesy of General Atomics)



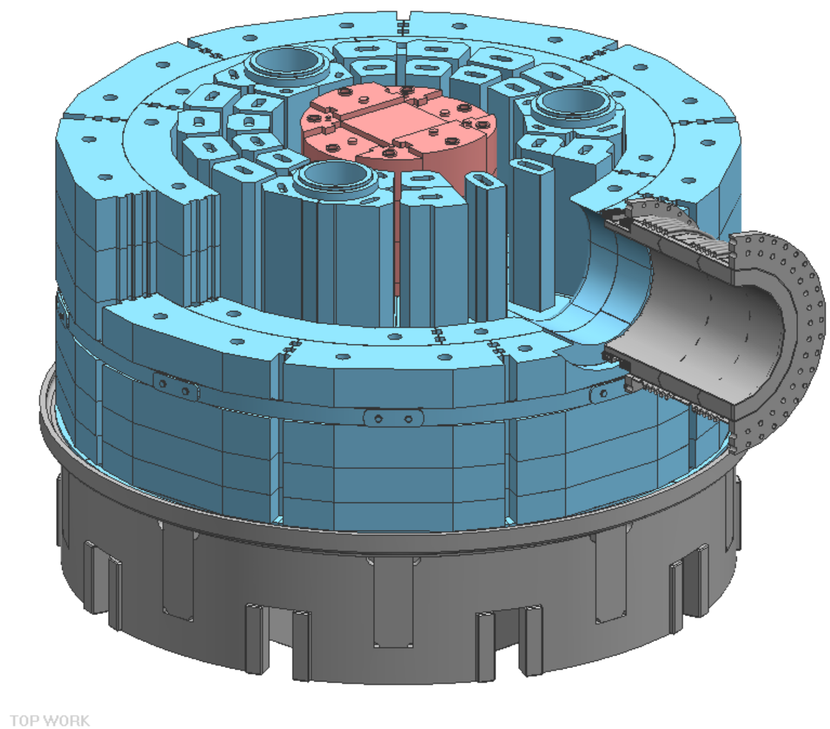
Pebble Bed HTGR (~ 800°C)

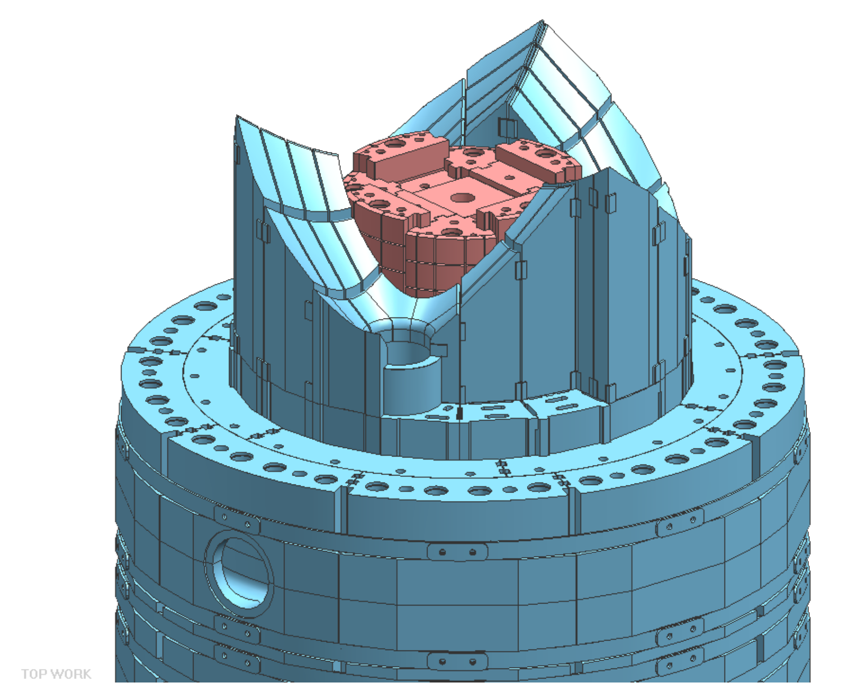
- Candidate materials:
  - Graphite
  - In -Vessel Structural CMC's (specifically C-C)
    - SIGRABOND 1501 YR
    - SIGRABOND 2001 YR
  - Control Rod (Alloy 800H to be replaced by Composite)
  - Vessel (Low Carbon Steel)
  - Insulation
    - Carbon
    - Fused quartz
    - Alumina



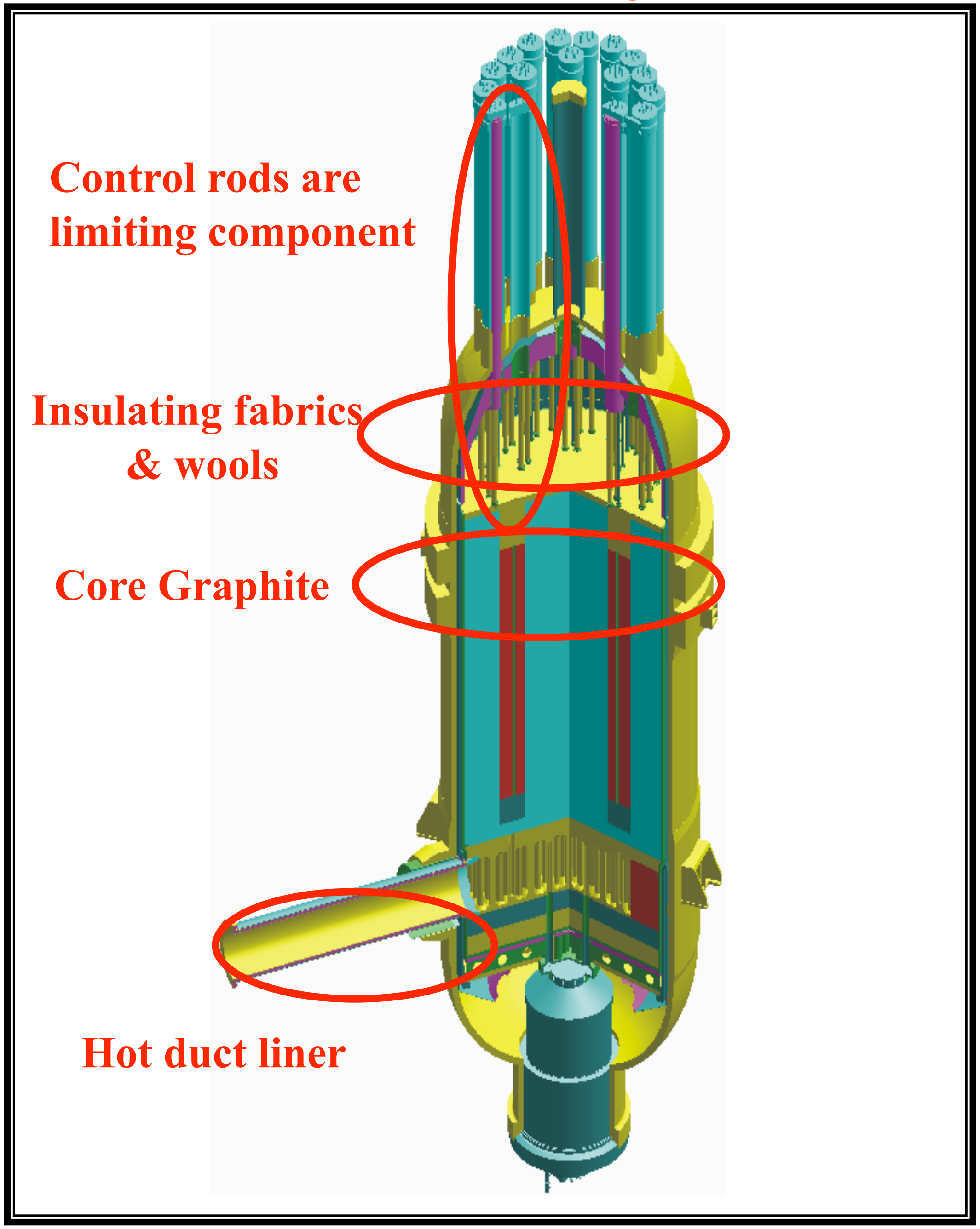
Courtesy PBMR, Pty Inc

Bottom Reflector

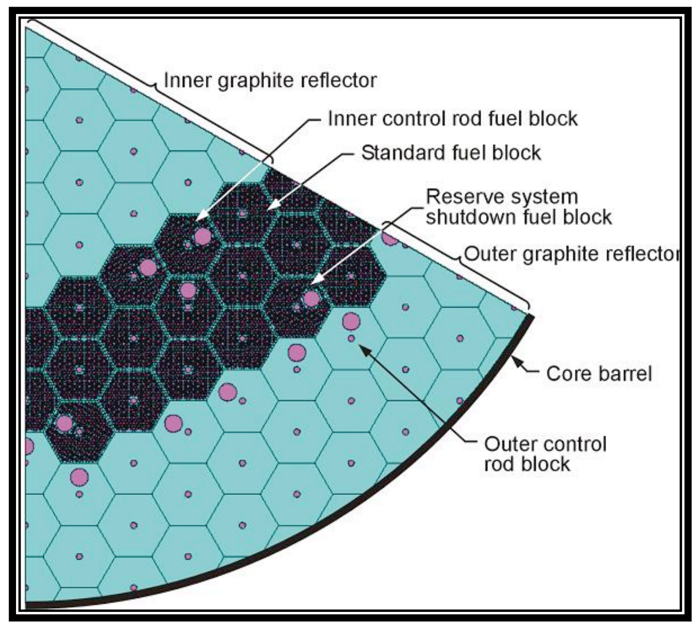




Components Operating > 1000°C



HTGR and VHTR



Outline

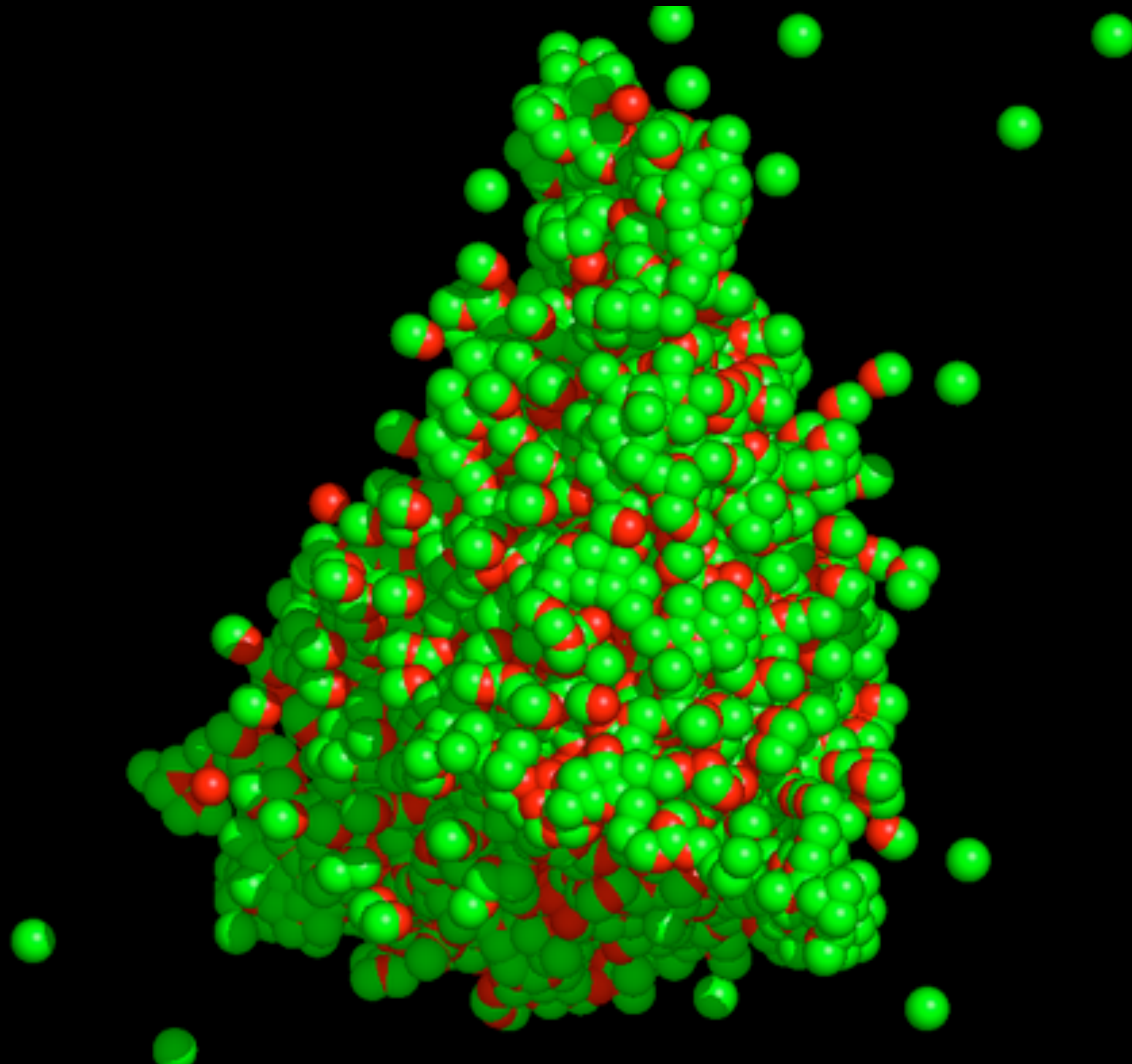
- **Brief Discussion of Gas Cooled Reactors**
  - Very early reactors and neutron moderators
- **Materials Issues for Key Components**
  - Pressure Vessel
  - Moderator
  - Control Rod
  - Internals

Defect Production and Annihilation in Irradiated Materials

-strong function of material, time, and temperature…

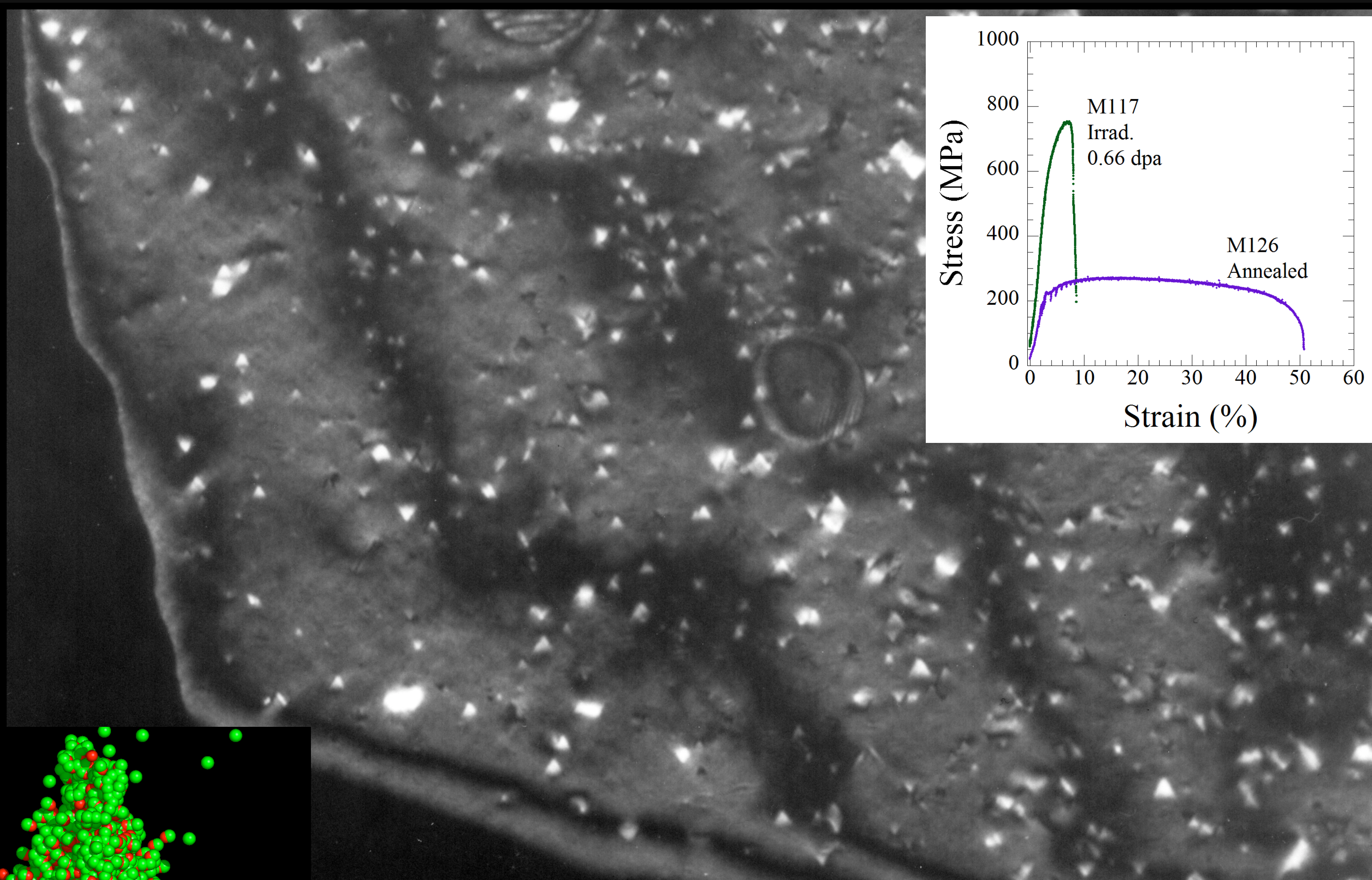
● vacancy ● interstitial

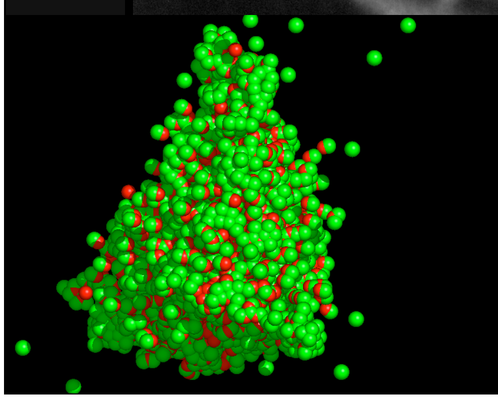
Condensation of Cascade in FCC Copper



A large non-perfect SFT and several SIA clusters/loops are formed

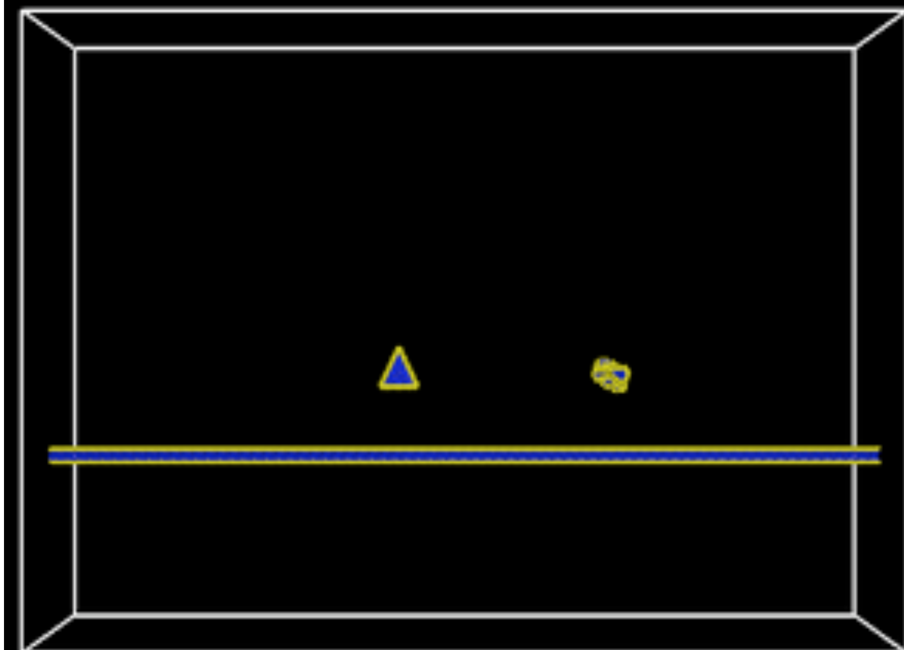
Defect cluster microstructure under irradiation 1 dpa at low temperature (~90˚C)

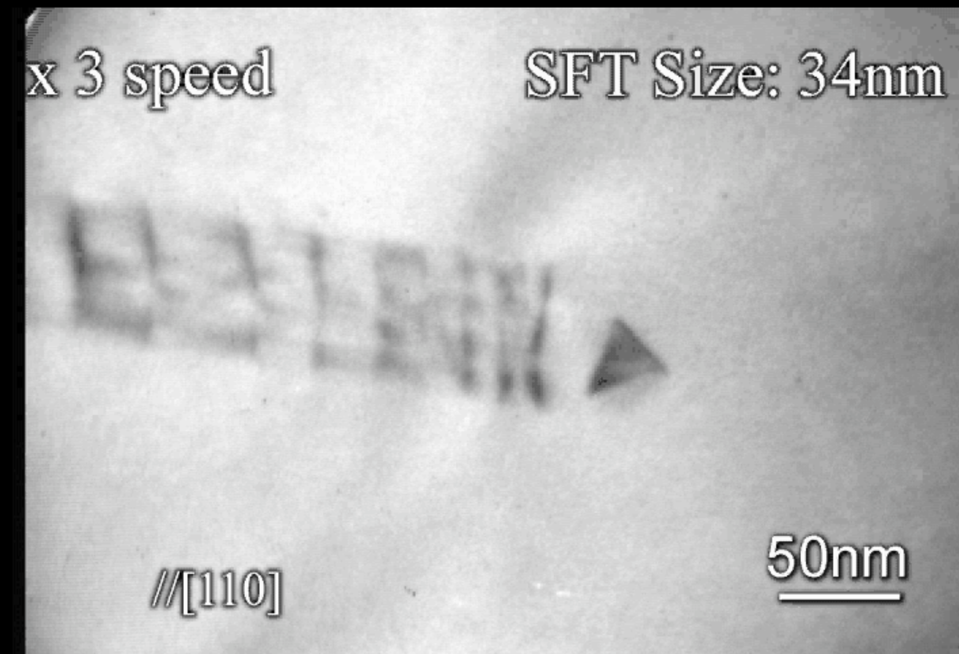




Zinkle et al., J. Nucl. Mater. (1994)

Dislocation dynamics and in-situ TEM of dislocation SFT interaction

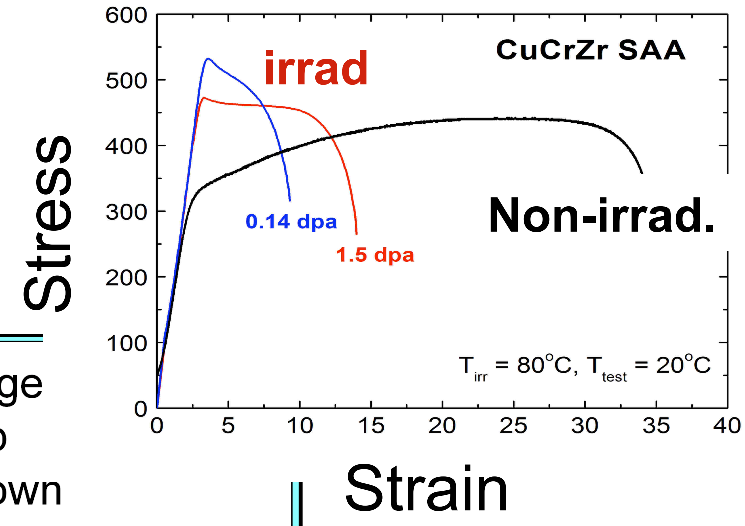


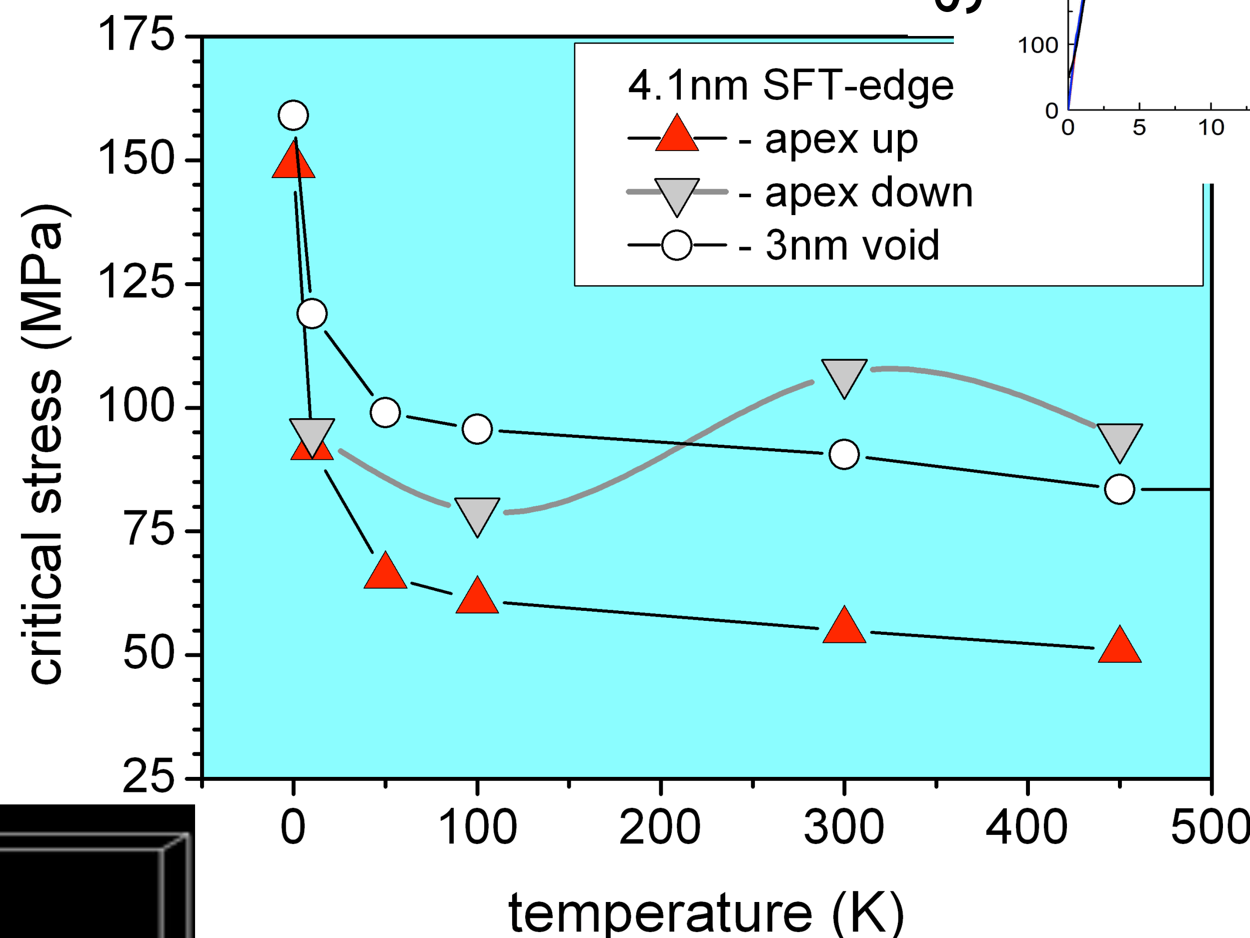


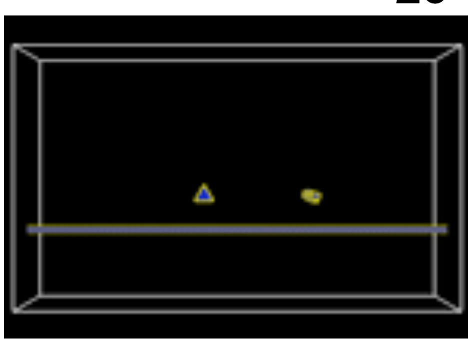
Yuri Osetskii, ORNL

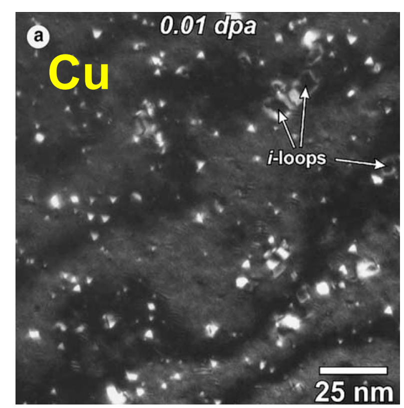
Yoshi Matsukawa,

Irradiation Hardening/Embrittlement







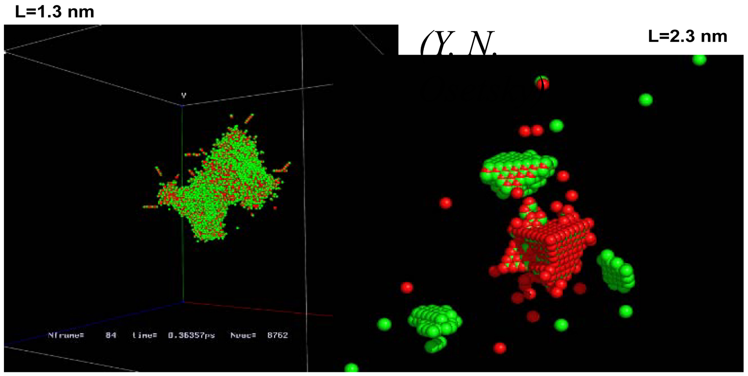


Irradiation Hardening and Embrittlement

- Irradiation produced defects serve as “road blocks” to the dislocation motion required for deformation (plasticity.)

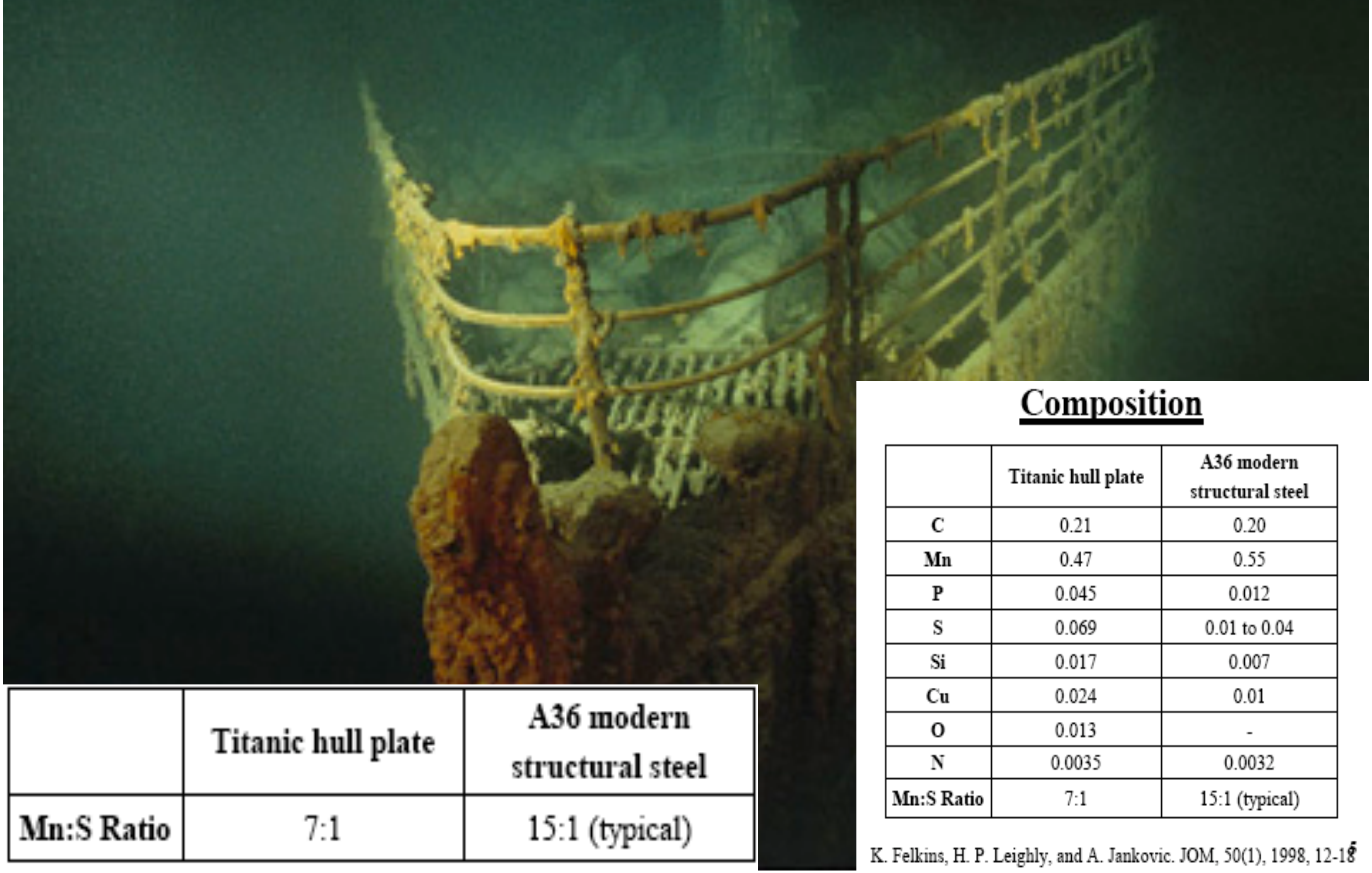
- Defects can be formed either directly within the cascade or can develop (mature) upon diffusion of interstitial/vacancy species following cascade:

Alloying and other metallurgical “tricks” can mitigate irradiation effects by tying-up or annihilating defects



- Irradiation Hardening
  - → increase in strength (yield and ultimate stresses)
  - → decrease in ductility
  - → increase in ductile-brittle transition temperature in Charpy test
  - → decrease in upper-shelf energy in Charpy test

Importance of Brittle to Ductile Transition Temperature
The RMS Titanic



	Titanic hull plate	A36 modern structural steel
Mn:S Ratio	7:1	15:1 (typical)

Composition

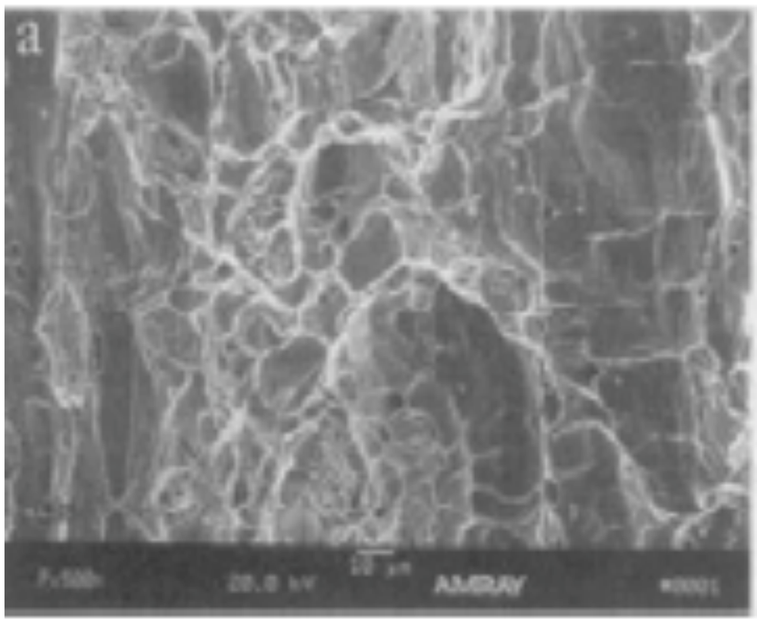
	Titanic hull plate	A36 modern structural steel
C	0.21	0.20
Mn	0.47	0.55
P	0.045	0.012
S	0.069	0.01 to 0.04
Si	0.017	0.007
Cu	0.024	0.01
O	0.013	-
N	0.0035	0.0032
Mn:S Ratio	7:1	15:1 (typical)

K. Felkins, H. P. Leighly, and A. Jankovic. JOM, 50(1), 1998, 12-18

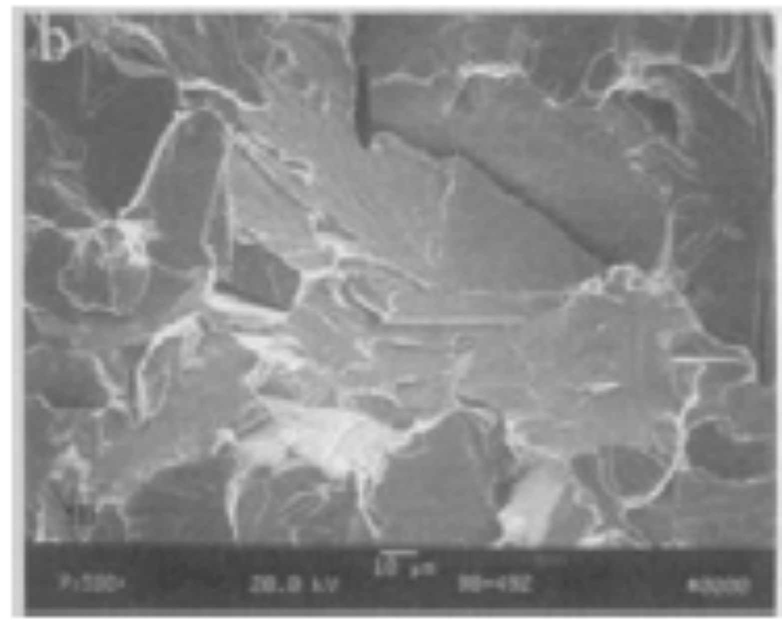
Importance of Brittle to Ductile Transition Temperature

Fracture surface of Charpy specimens from Titanic plate

Longitudinal direction

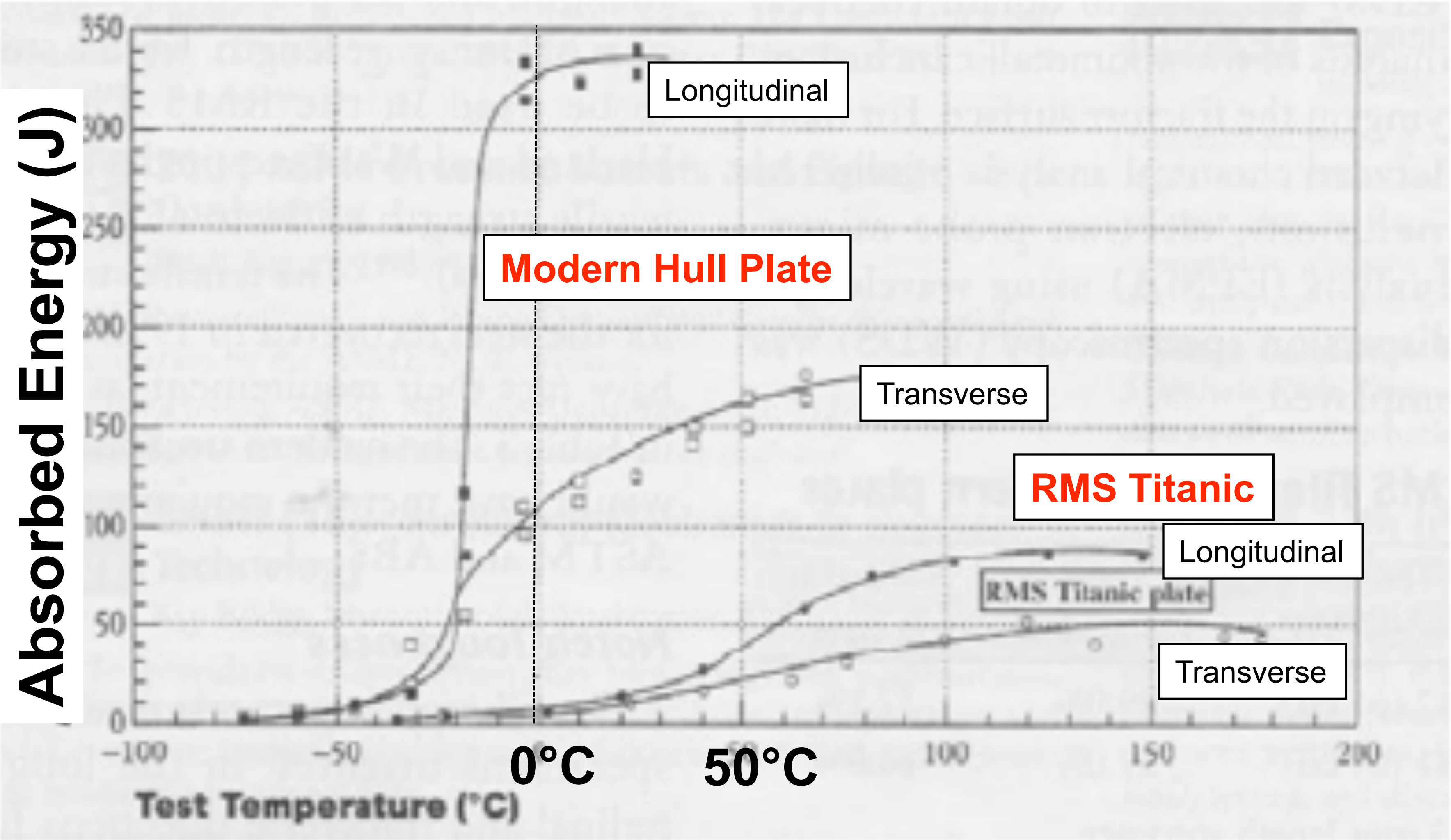


At 120 °C,
ductile fracture



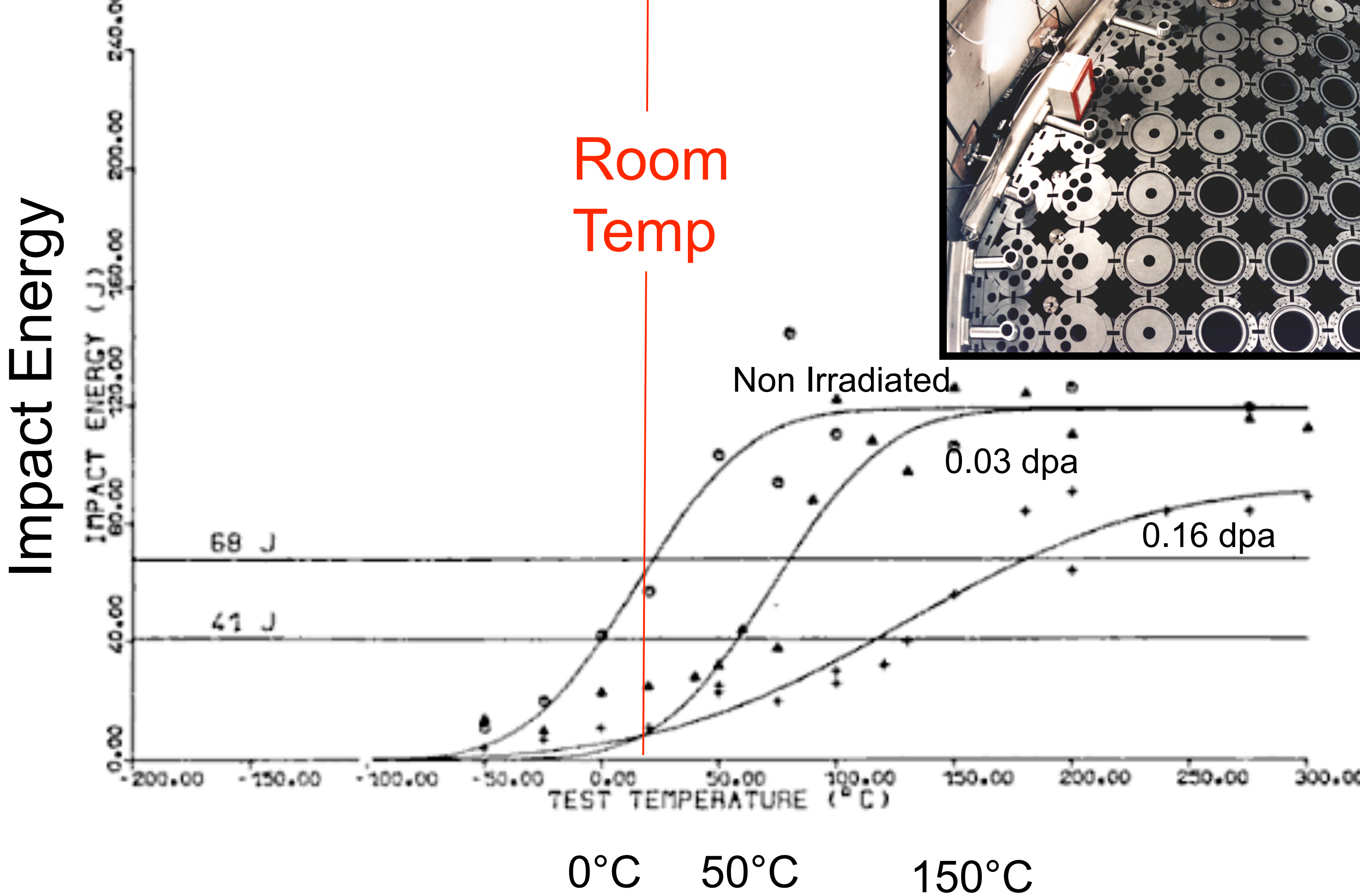
At -32 °C,
brittle fracture

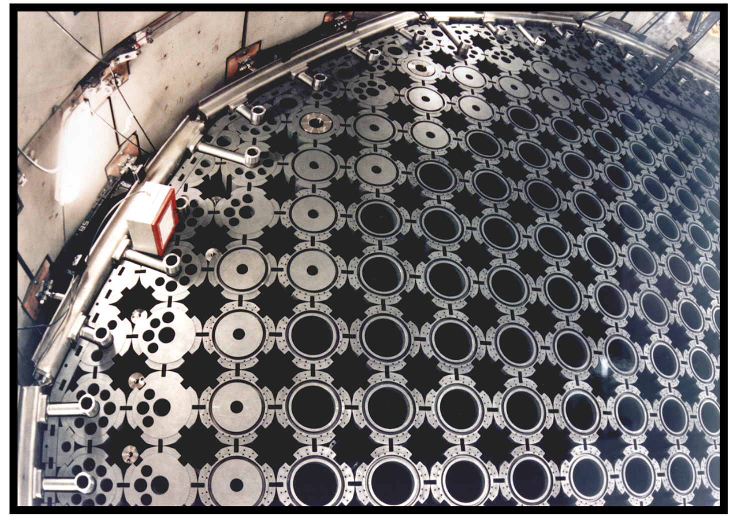
Effect of temperature on toughness



H. P. Leighly, B. L. Bramfitt, and S. J. Lawrence. Practical Failure Analysis, 1(2), 2001

AGR Irradiated Pressure Vessel Steel



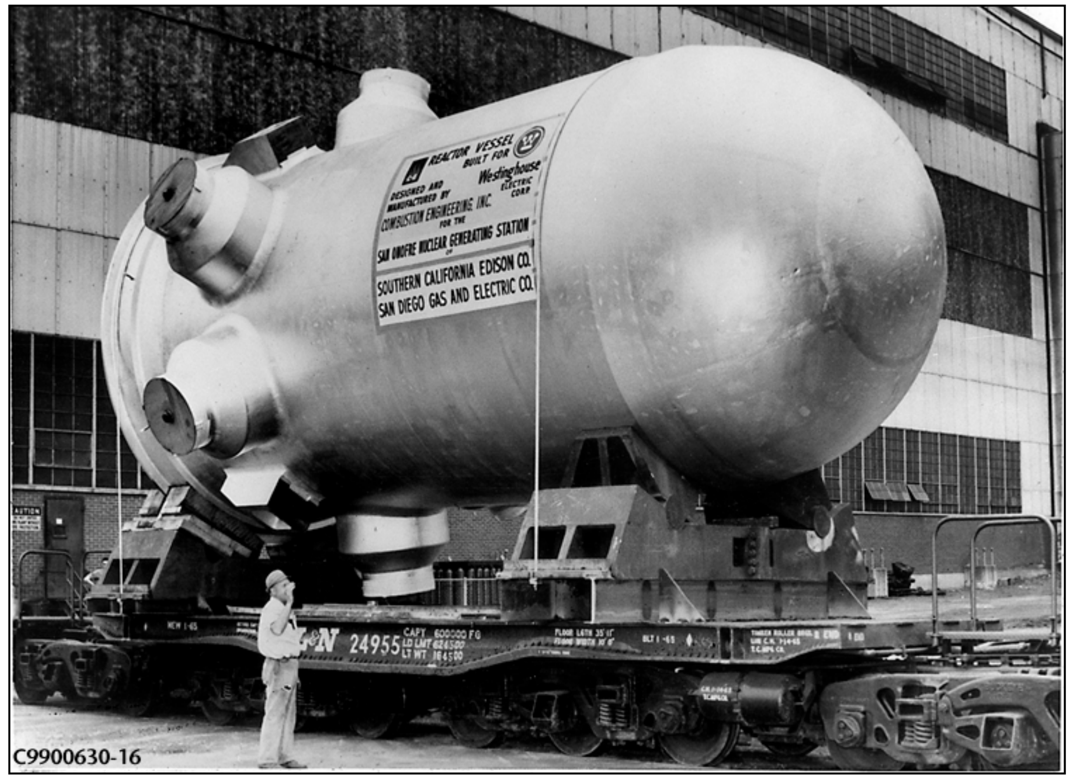


0.19 C, 0.3 S, 1.22 Mn, 0.07 S, 0.015 P, 0.12 Cu, 0.11 Ni, 0.1 Cr, 0.03 Mo, 0.01 Al

Reactor Pressure Vessels For Modern Day Nuclear Power Plants Are Complex Heavy-Section Steel Components

- Vessels may weigh up to 800t with wall thickness up to ~330mm (~13 in.)

- Cylindrical and hemispherical sections of quenched and tempered low-alloy steel (Mn-Mo-Ni) plates or forgings are welded together, usually by submerged-arc welding, clad on the inside with stainless steel weld metal, and given a final postweld heat treatment



History of reactor pressure vessel material developments for fission reactors have been stagnant for the past 40 years.

• A few early reactor pressure vessels for light water reactors in the early 1960s were fabricated from A212 grade B steel, essentially a carbon steel (~0.2 wt%) with some alloying elements to provide a modest amount of hardenability; the specified yield strength was about 310 MPa (45 ksi.)

• Subsequently, SA-302, Grade B, a low-alloy manganese-molybdenum plate steel with a specified minimum yield strength of 345 MPa (50 ksi), was used for a number of vessels made through the mid-1960s and, as commercial nuclear power evolved, the sizes of the vessels increased, requiring greater wall thickness and a material with greater hardenability.

• The addition of about 0.5 wt% nickel to SA-302 grade B provided the necessary increased hardenability to achieve the desired yield strength and high fracture toughness across the entire wall thickness, but the specified yield strength was not higher. This general composition and strength level became the current steels (Fe-0.2C-0.6Ni-1.2Mn-0.5Mo), SA 533 grade B class1 plate and SA 508 class 2/ class 3 forgings, which have been used for all PWRs and BWRs since the early 1970s.

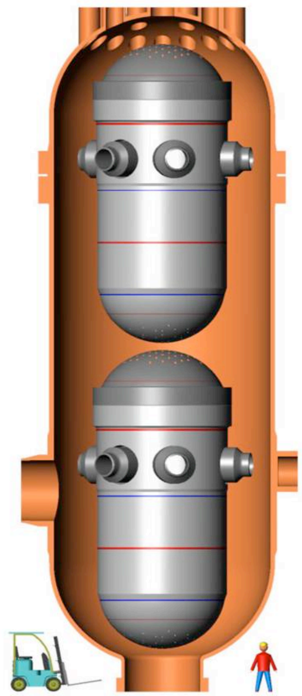
History of reactor pressure vessel material developments for light water reactors have been stagnant for the past 40 years (Cont'd)

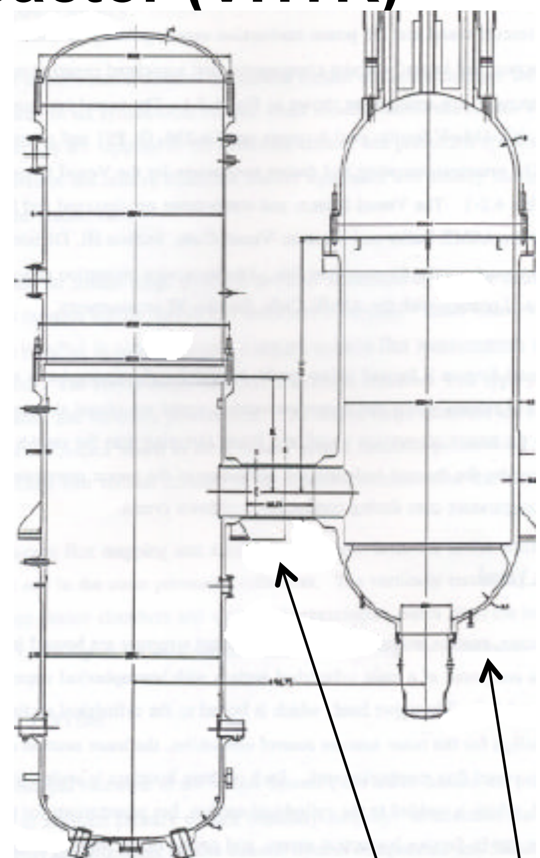
- RPV materials research for the past 40 years has focused on development of fracture mechanics and understanding of radiation embrittlement mechanisms, with no development of new steels.

- There are non-commercial RPVs constructed of A508 grade 4N class 1 (a low alloy steel but with ~ 3.5 wt% Ni), which has a minimum specified yield strength of 550 MPa (80 ksi), allowing for a substantial reduction in thickness for a given RPV size, but the commercial nuclear industry has not made moves towards use of higher strength steels for new LWRs, including a new steel, 3Cr3WV, with high hardenability.

For the Generation IV Reactor Program, 9Cr-1MoV Is Primary Choice for the Reactor Pressure Vessel in the NGNP Very High Temperature Reactor (VHTR)

- Up to 490ºC (previous = 650°C) operating temp and 3×10^{19} n/cm^2 fluence (>0.1 MeV), 0.0075dpa
- Issues include irradiation effects in creep range, and long-term strength
- High-temperature design methodology needs updating for nuclear service
- Very large vessel sizes will require scale-up of ring forging and joining technologies and ensuring thick-section properties
- *More advanced ferritic steels, e.g., 9Cr-2WV and 3Cr-3WV, have been considered as well.*





Two typical PWR RPVs fit inside NGNP RPV

PCV

Cross vessel

RPV

Considerations for Application of Advanced Steels for High-Temperature Reactor Pressure Vessels

- For highly pressurized systems, like the VHTR, very large pressure vessels are required. They must have sufficient hardenability for thick section fabrication as well as relatively high strength at temperatures in the range of 600 to 700°C, and resistance to neutron radiation.
- The well-known Grade 91 steel, 9Cr1MoV, is approved in the ASME Code for high temperature application. There are other advanced options with better high-temperature strength, but they are not qualified for ASME Code application.
- Steels such as reduced activation ferritic-martensitic steels (RAFS), exemplified by F82H, JLS-1, Eurofer 97 and 9Cr-2WVTa, have been developed in recent years, but without Code qualification.
- The next generation of steels is being developed at present, where the intention is to push operating temperatures to 650ºC. These fourth-generation steels, SAVE12 (Fe-11Cr-3W-3Co-0.20V) and NF12 (Fe-11Cr-2.6W-2.5Co), differ from the previous generation primarily by the use of up to 3.0% cobalt; they have projected 105 h creep-rupture strengths at 600ºC of 180 MPa

Considerations for Application of Advanced Steels for High-Temperature Reactor Pressure Vessels (Cont'd)

- Other developments to be pursued involve the use of conventional steelmaking techniques to produce a high number density of fine precipitates into the steel matrix. Some success has been achieved in producing steels that contain a high number density of small nitride precipitates. This was accomplished by a thermo-mechanical treatment, wherein hot working was introduced into the normal heat treatment sequence typically used for conventional steels. The thermo-mechanical treatment produces a high number of dislocations into the matrix, and these dislocations act as heterogeneous nucleation sites for the nitride precipitates which could also provide relatively high radiation resistance.
- **These steels could also push the use temperature well above the 600-620ºC limit of present high-temperature ferritic steels.**
- The ability to use conventional steelmaking techniques means that relatively thick sections could be manufactured, a necessary requirement for many structural components like pressure vessels.
- **Code qualification of such advanced steels is likely to take at least 10 years,** given adequate funding for all the necessary fabrication of thick sections, welding, etc.

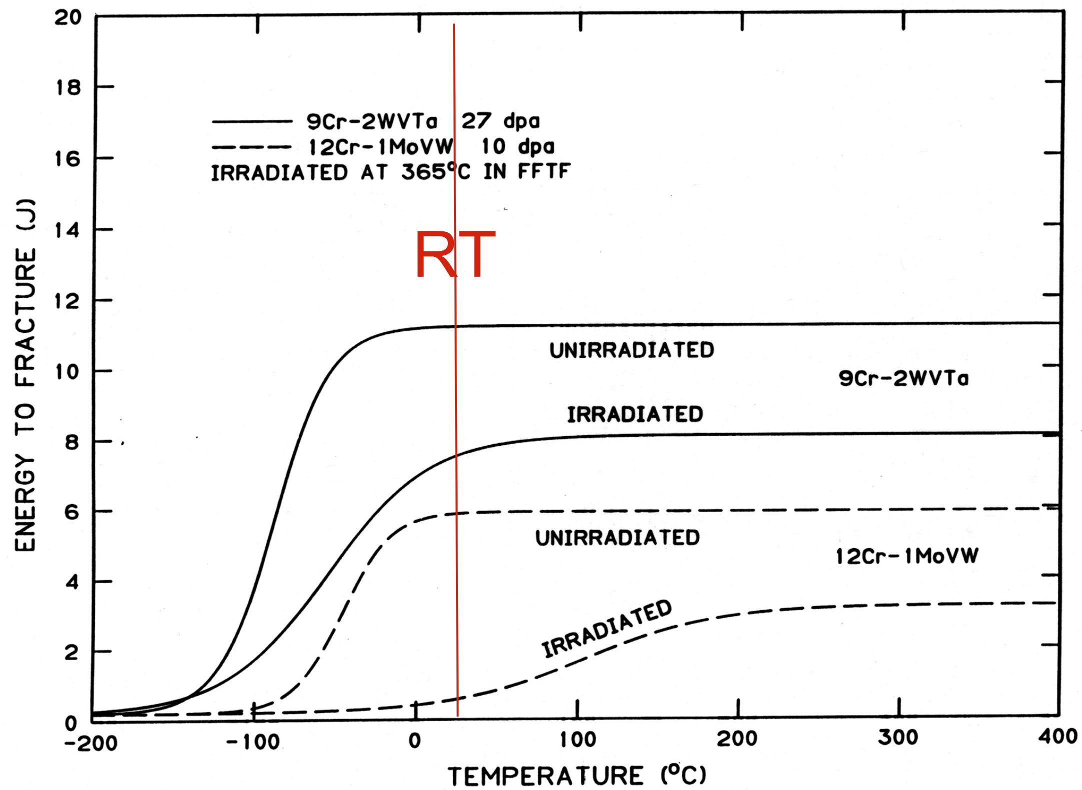
Evolution of Steels For Power-Generation Industry

Generation	Years	Steel Modification	10^5 h Rupture Strength, 600°C (MPa)	Steels	Max Use Temperature (°C)
0	1940-60		40	T22, T9	520-538
1	1960-70	Addition of Mo, Nb, V to simple Cr-Mo steels	60	EM12, HCM9M, HT9, HT91	565
2	1970-85	Optimization of C, Nb,V, N	100	HCM12, T91, HCM2S	593
3	1985-95	Partial substitution of W for Mo and add Cu, B	140	NF616, E911, HCM12A	620
4	Future	Increase W and add Co	180	NF12, SAVE12	650

Irradiated Fracture Behavior of Newer Generation Steels

- •9Cr-2WVTa superior to HT9(12Cr-1MoVW)
- •Mod 9Cr-1Mo superior to HT9
- •9Cr-2WVTa superior to mod 9Cr-1Mo

HT9 vs. 9Cr-2WVTa



- The Newer Generation steels have good elevated temperature strength and irradiated "transition temperatures" above room temperature. With appropriate effort in code qualification, these materials are quite promising.

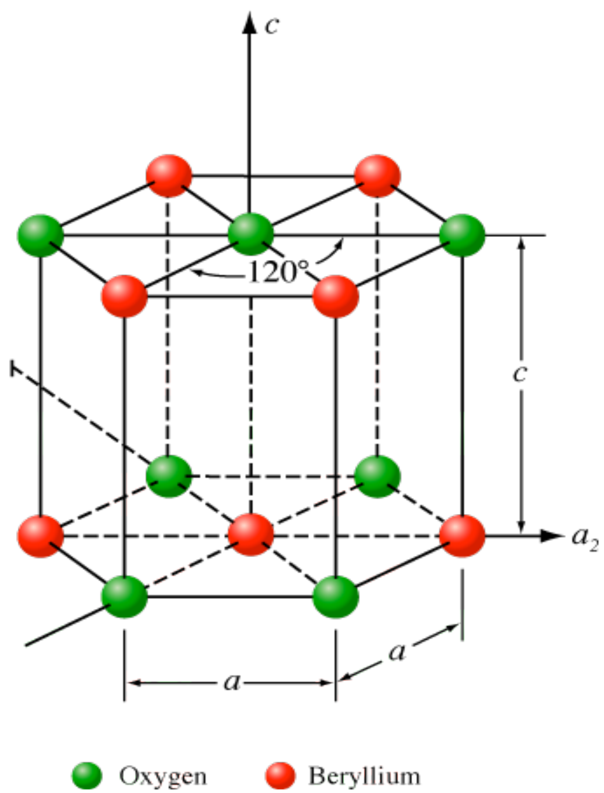
Neutron Moderators

- **Early piles and gas cooled reactors focused on two “purified” ceramics with very low neutron absorption cross sections, BeO and graphite….**

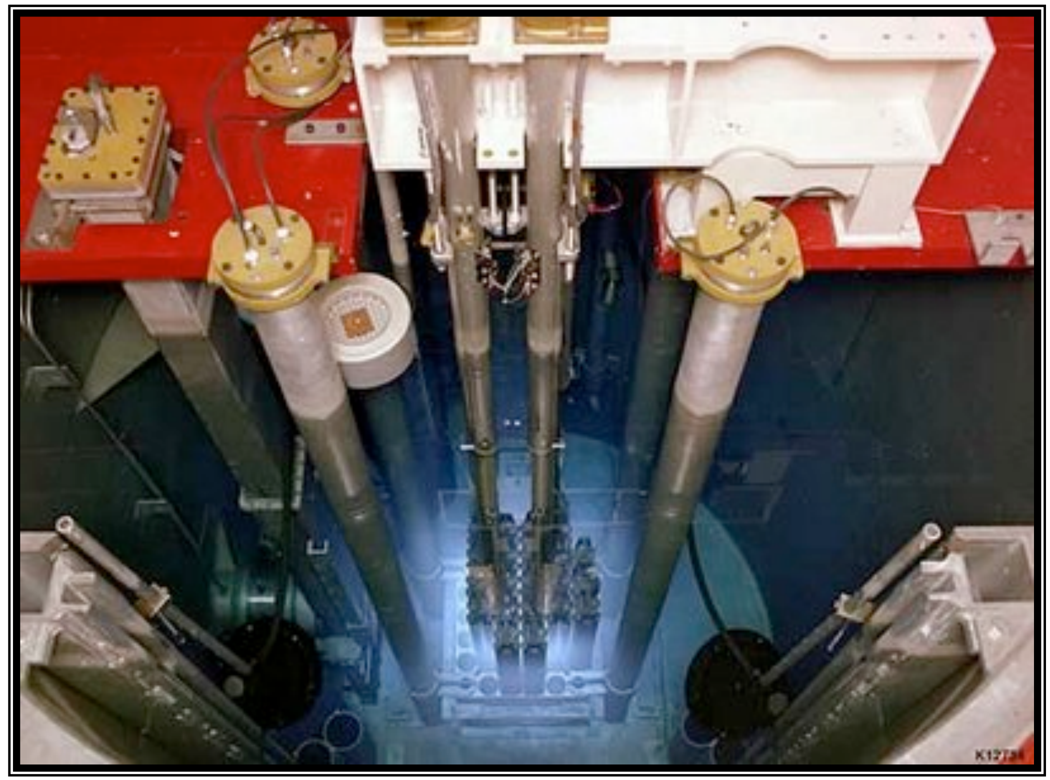
Neutron Cascade Damage and Defect Evolution

The BeO Crystal

Irradiation-induced defects produce anisotropic swelling within the hexagonal structure of the BeO

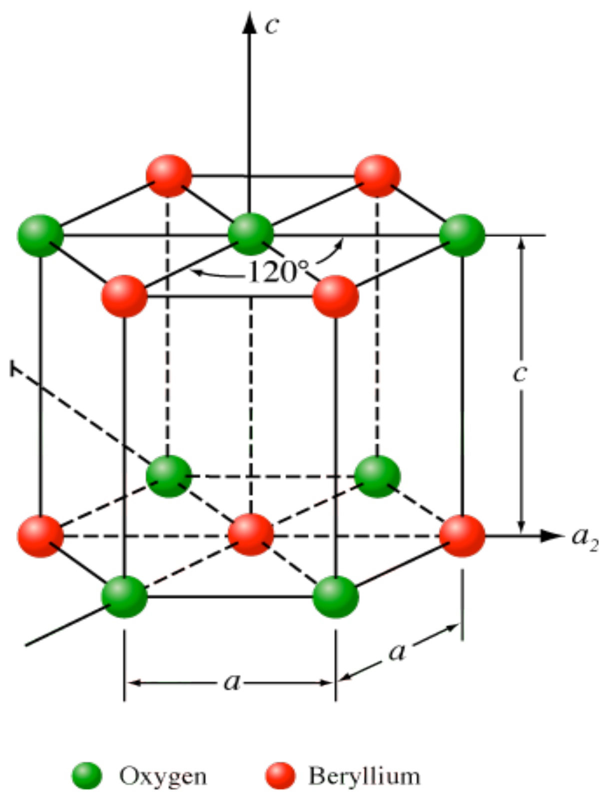


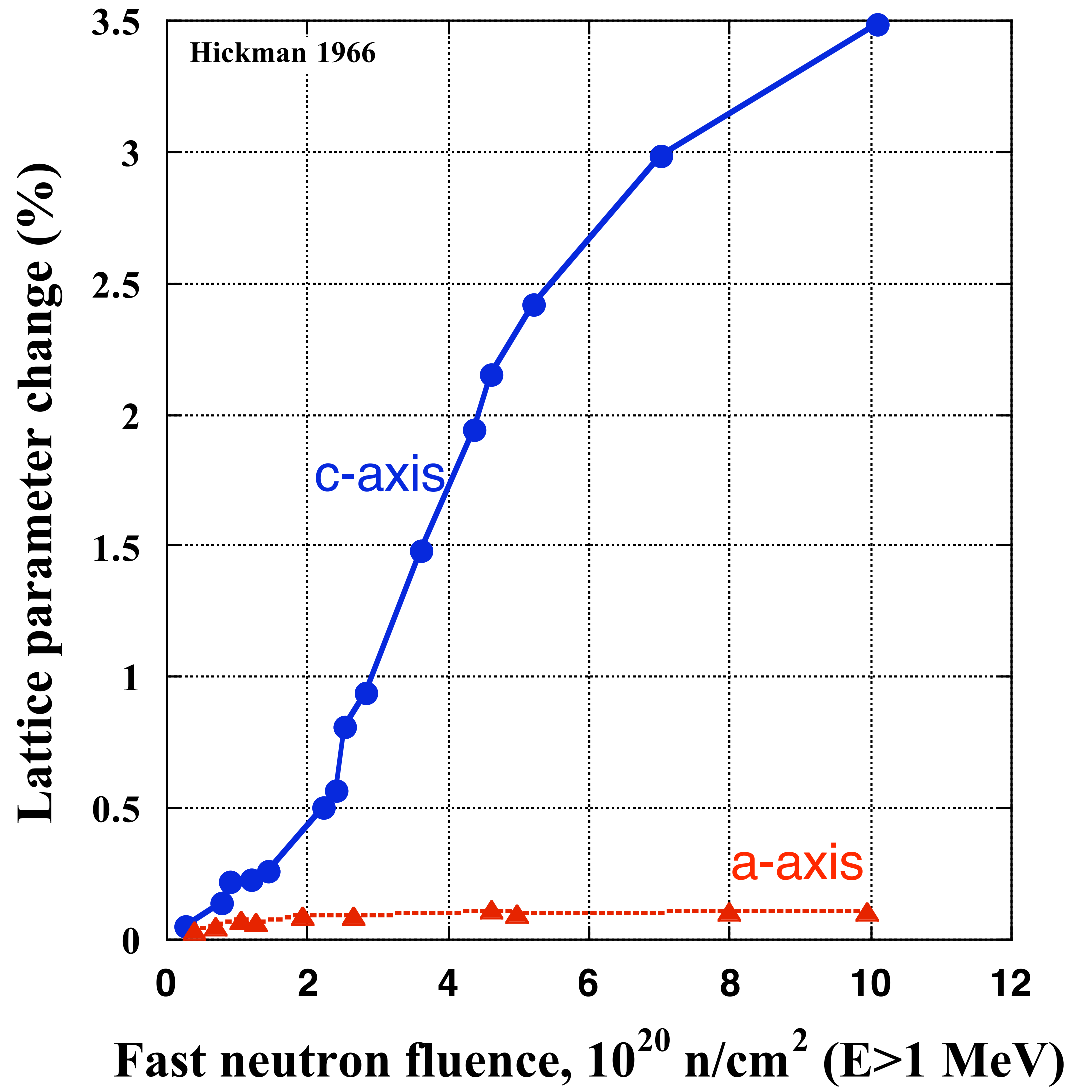
TRIGA Reactor



The BeO Crystal

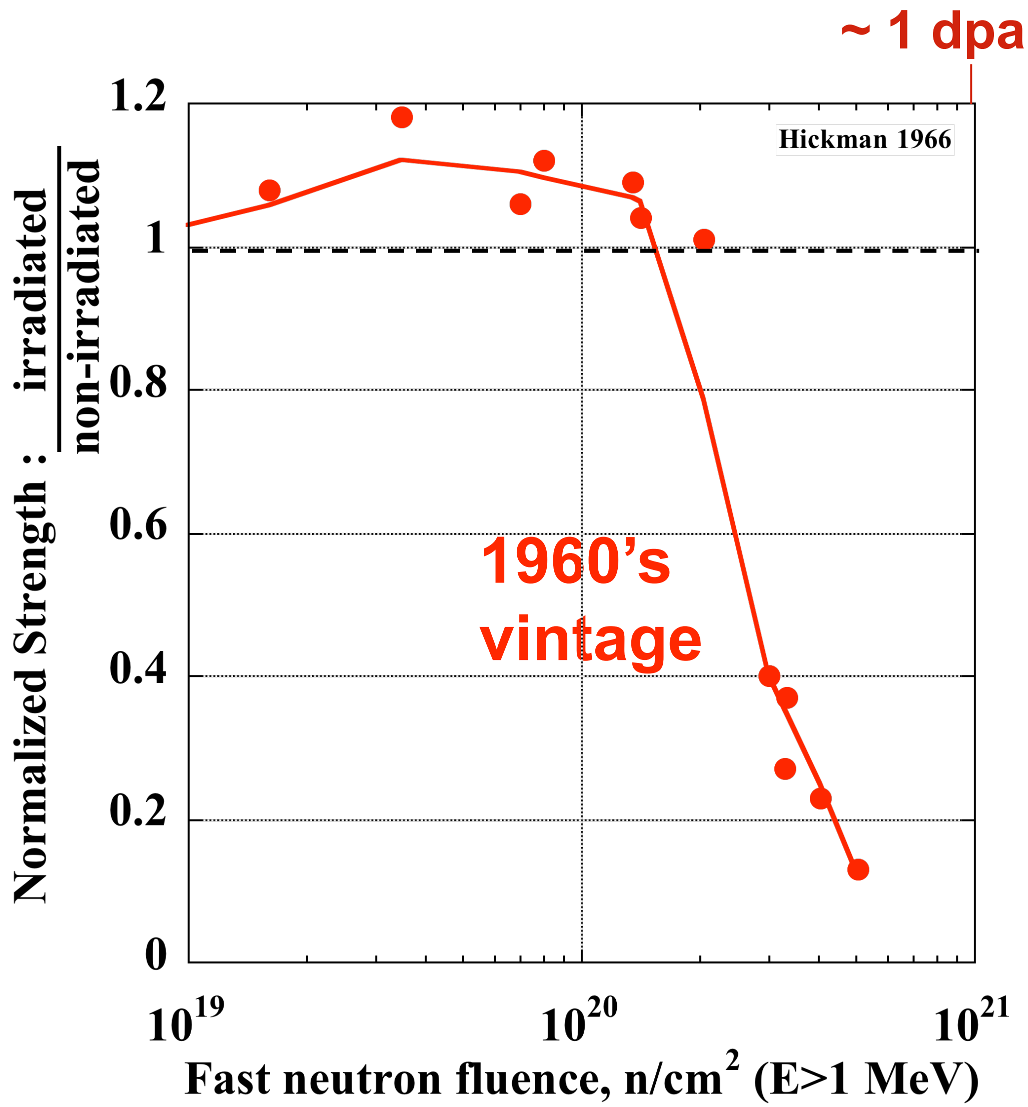
Irradiation-induced defects produce anisotropic swelling within the hexagonal structure of the BeO





Effect of Neutron Irradiation on BeO Strength

- Anisotropic swelling produces strain with crystal
- Strain causes eventual rupture of grain boundaries
- Such rupture is unavoidable, limiting useful life of BeO
- Modern BeO is somewhat stronger with stronger grain boundaries. A small improvement in irradiation performance is expected.



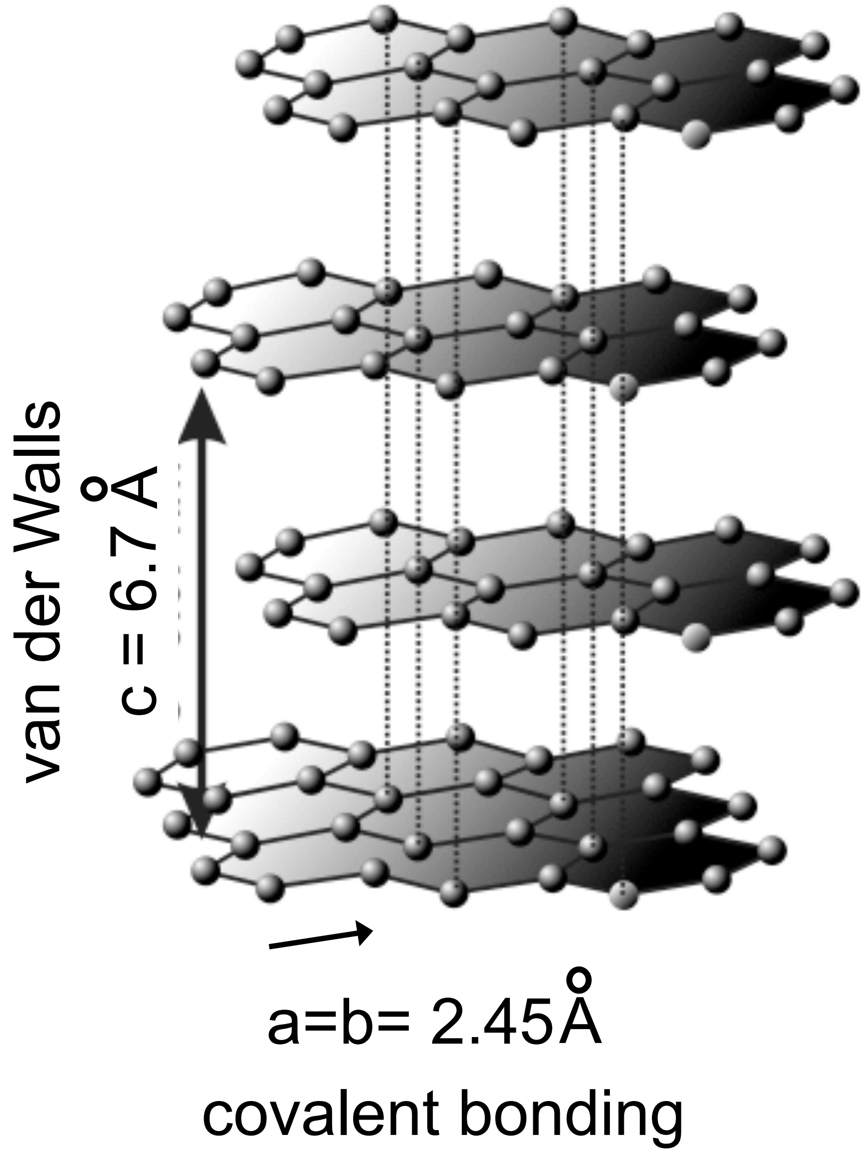
The Graphite Crystal

- The graphite crystal is an interpenetrating hexagonal "benzene" ring structure.

- Very weak bonding between planes, strong bonding in planes.

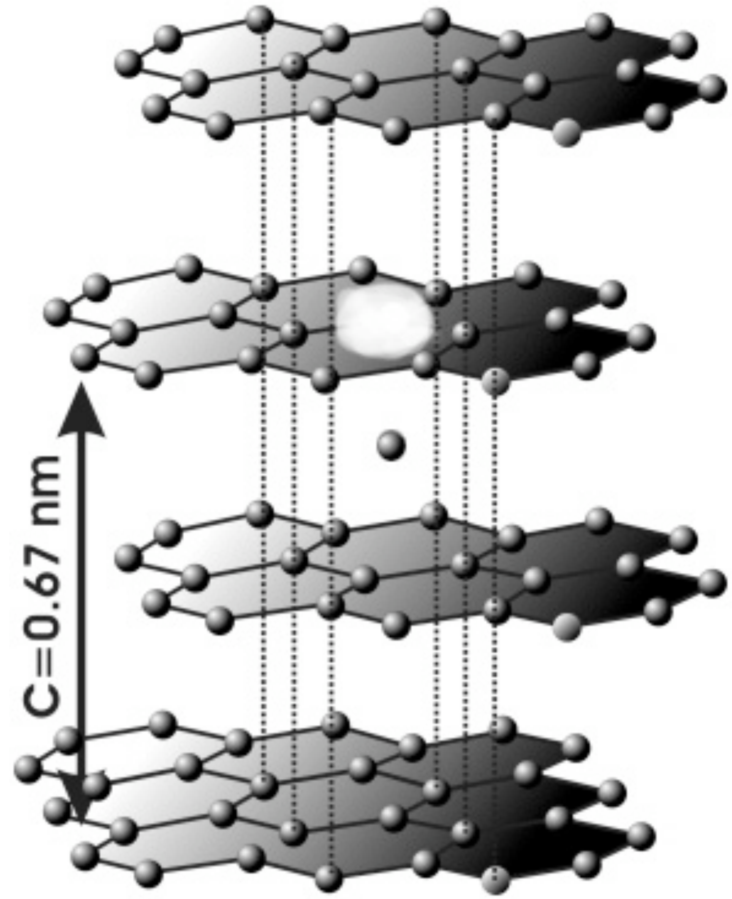
- Extraordinary in-plane properties, drastically different out-of-plane.

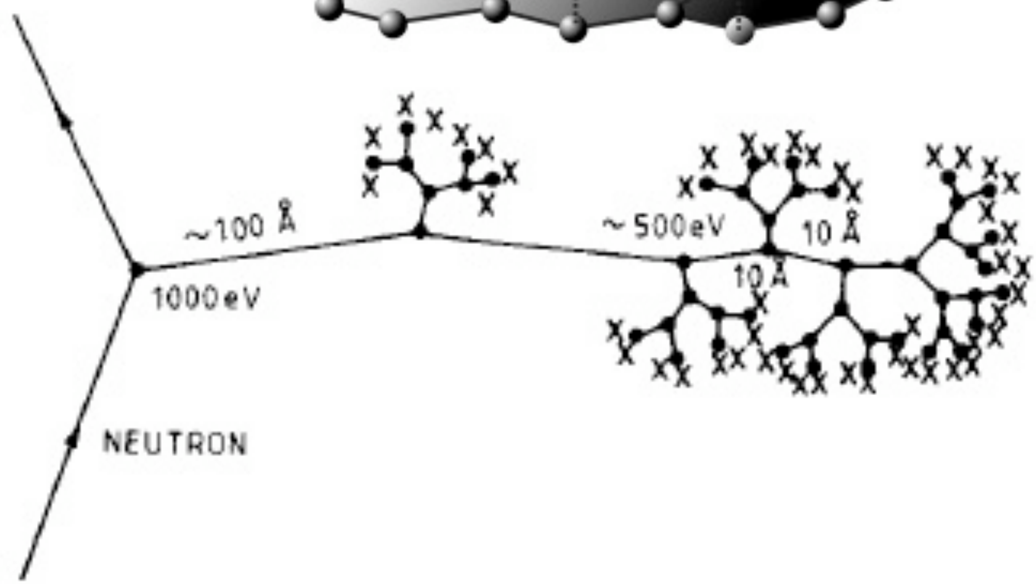
	In-plane	Out-plane
Thermal Conductivity W/m-K)	>2200 W	20
Thermal Expansion	0.5	6.5
Strength (MPa)	>1000	<1
Elastic Modulus(GPa)	20	<1



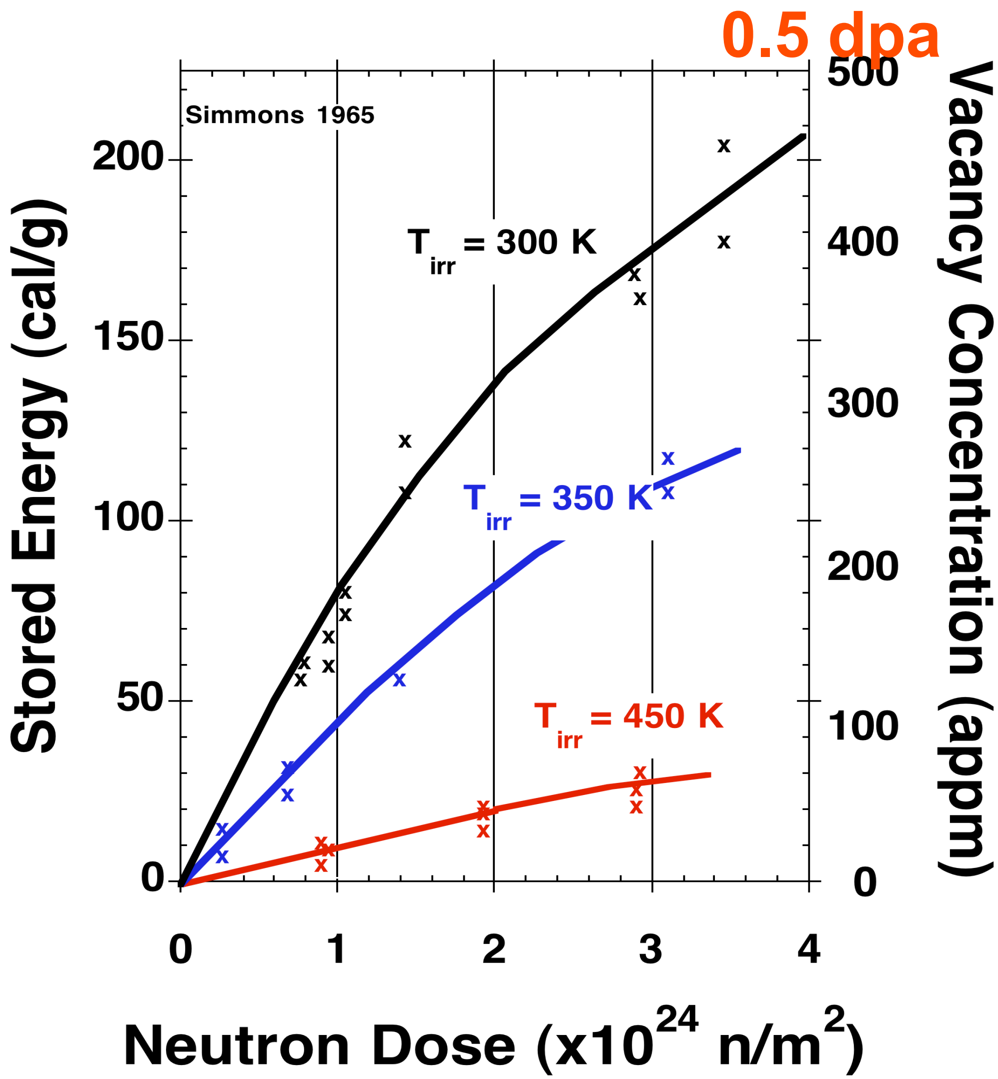
Irradiated Graphite Crystal

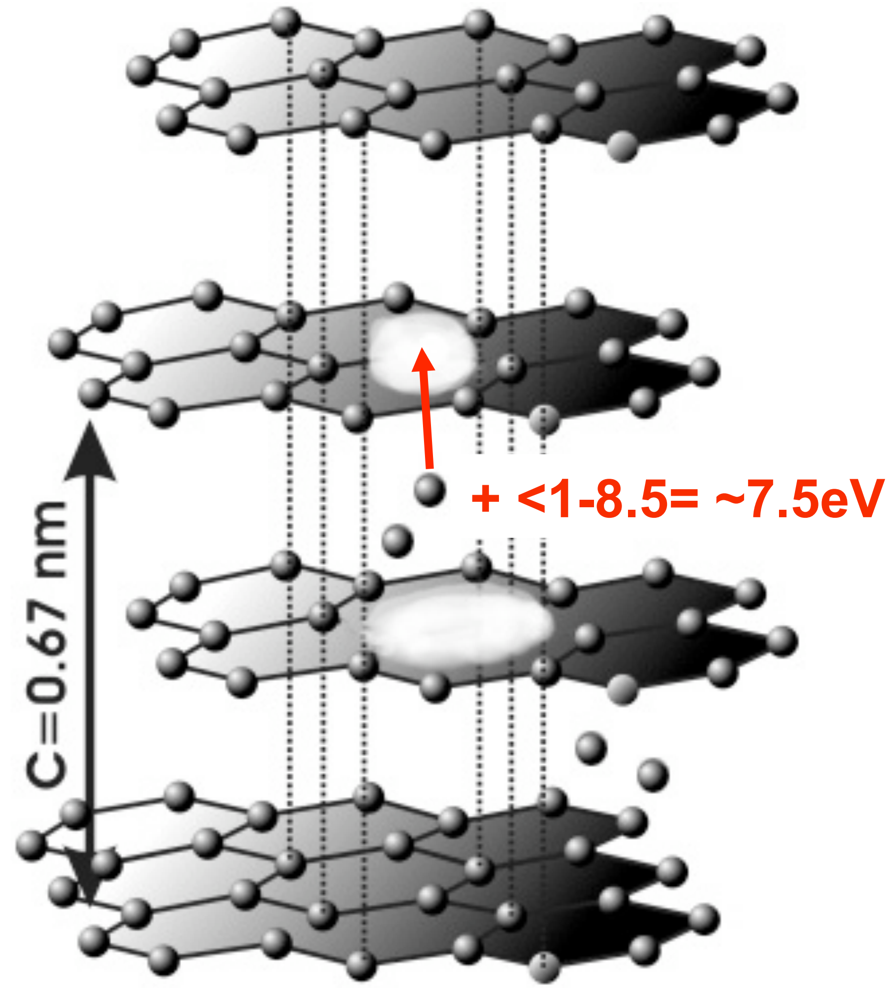
- Interstitials mobile > 70 K, move within the basal plane.
- Vacancies mobile > 1000 K, move freely between the basal plane.
- Interstitial-vacancy recombination barrier < 1 eV (<400K).





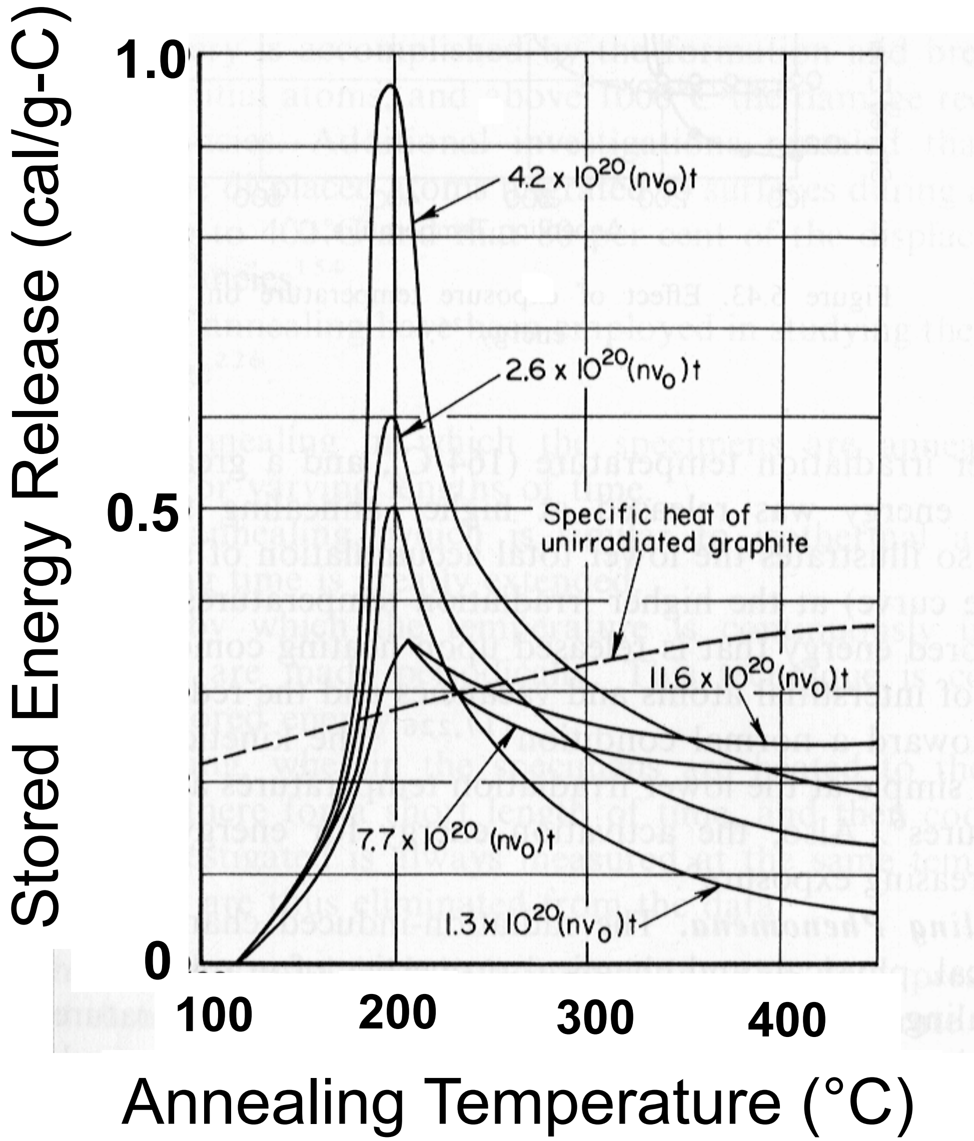
Simple Defect Recombination in Graphite : Low Temperature





- As irradiation temperature exceeds activation energy for recombination, vacancy concentration and stored energy is reduced

Simple Defect Recombination in Graphite : Low Temperature



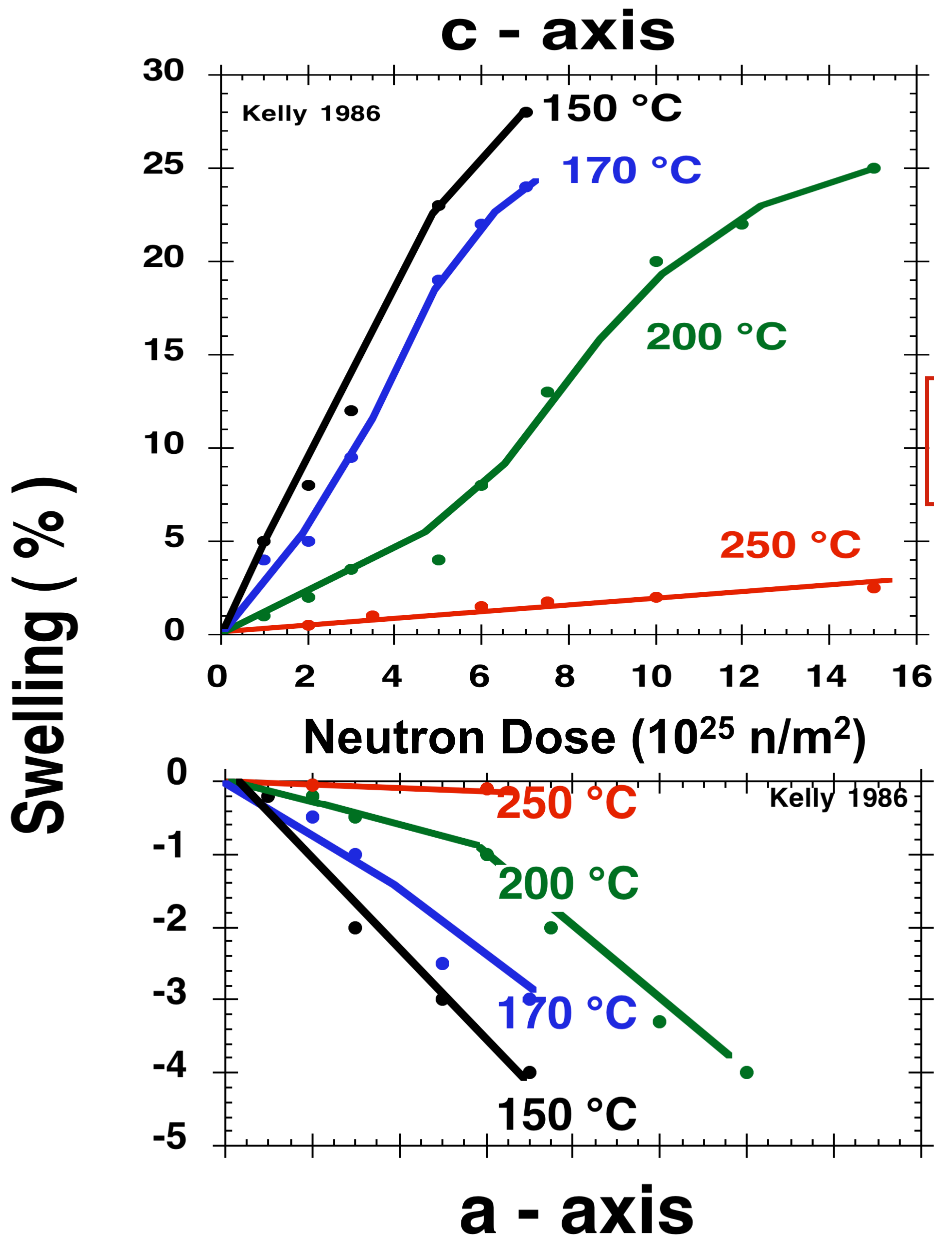
Windscale

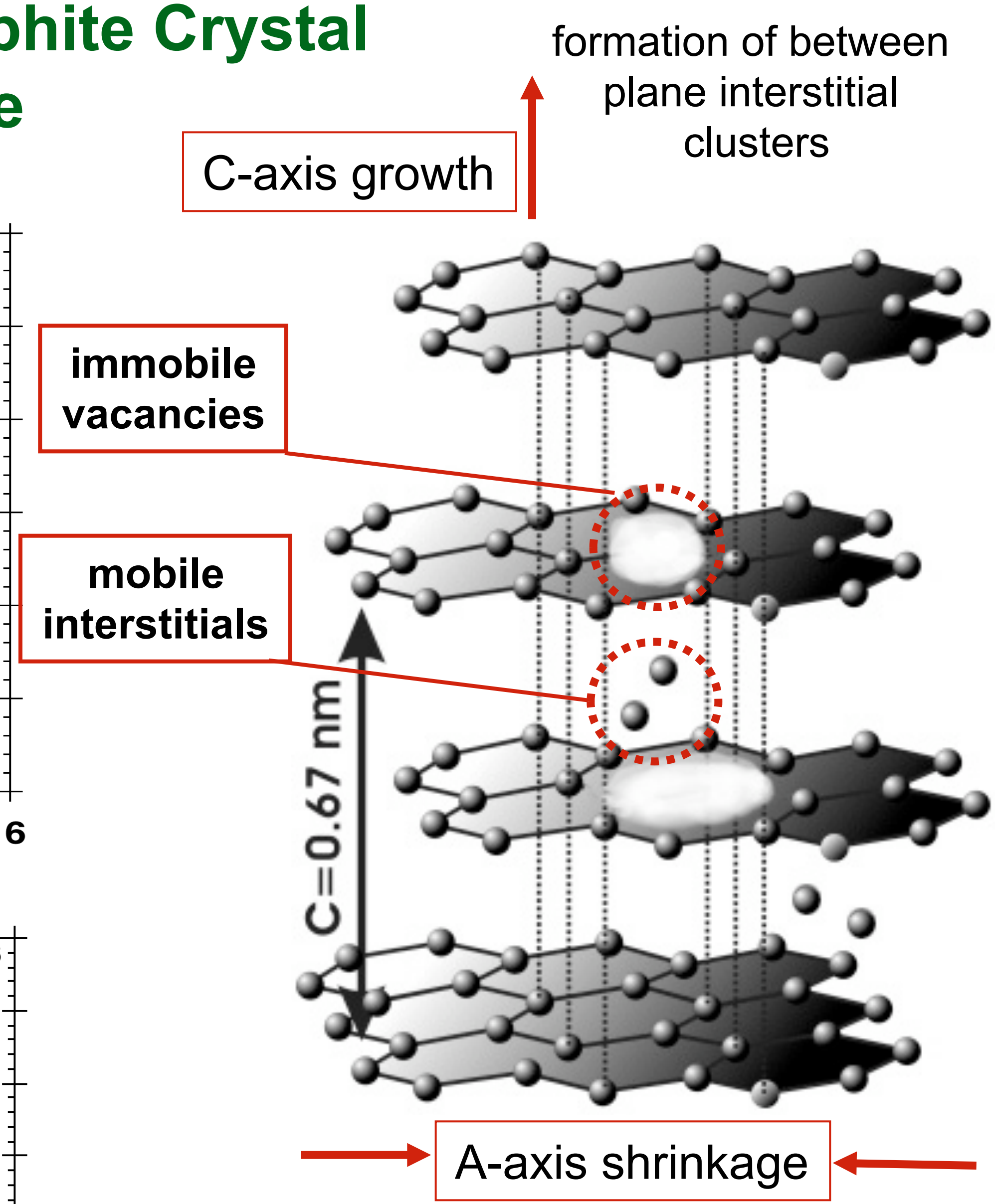
Air cooled, Graphite Moderated

Operating Temperature ~ 250°C

Core burned to > 1300°C for five days

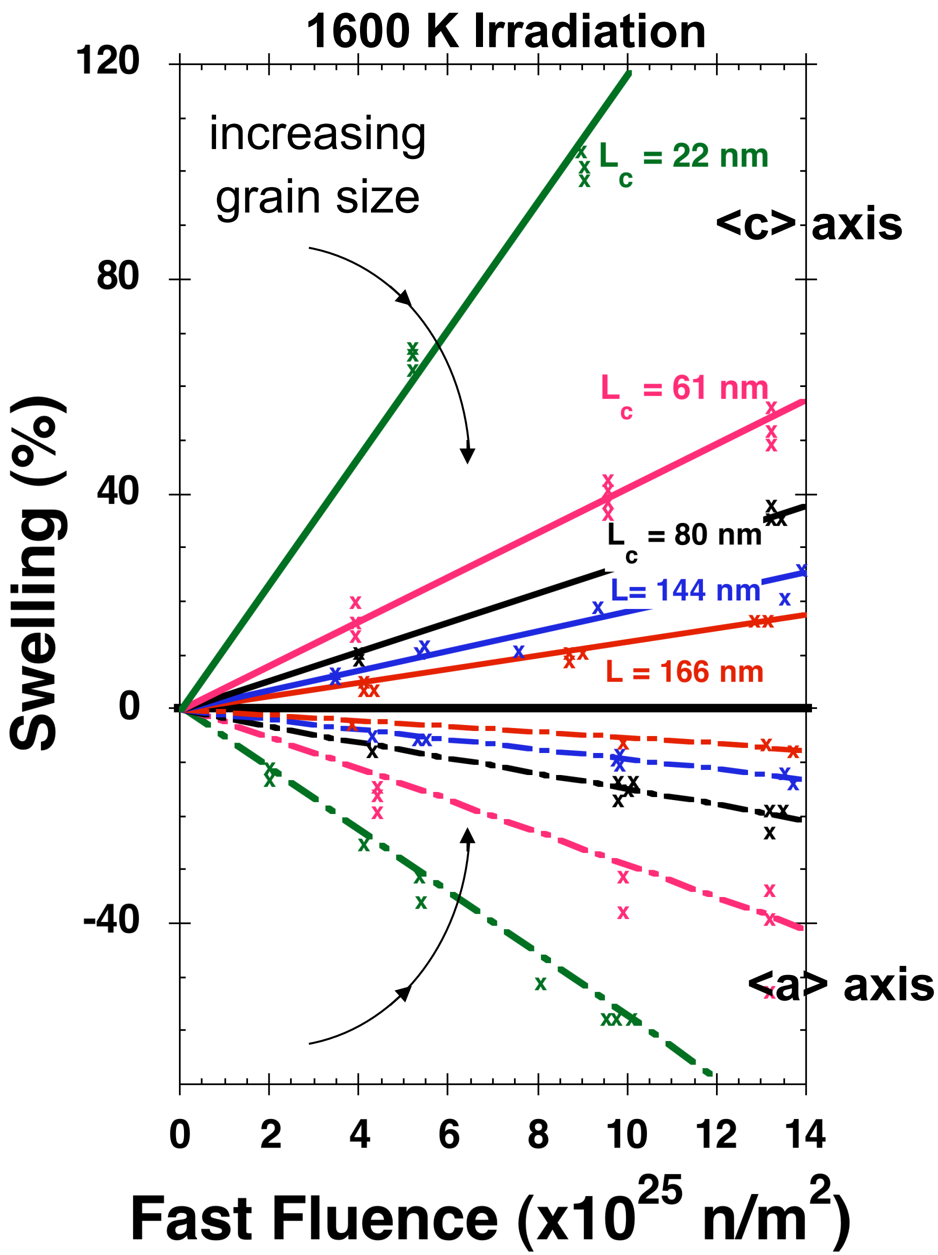
Dimensional Change in Graphite Crystal
Low Temperature





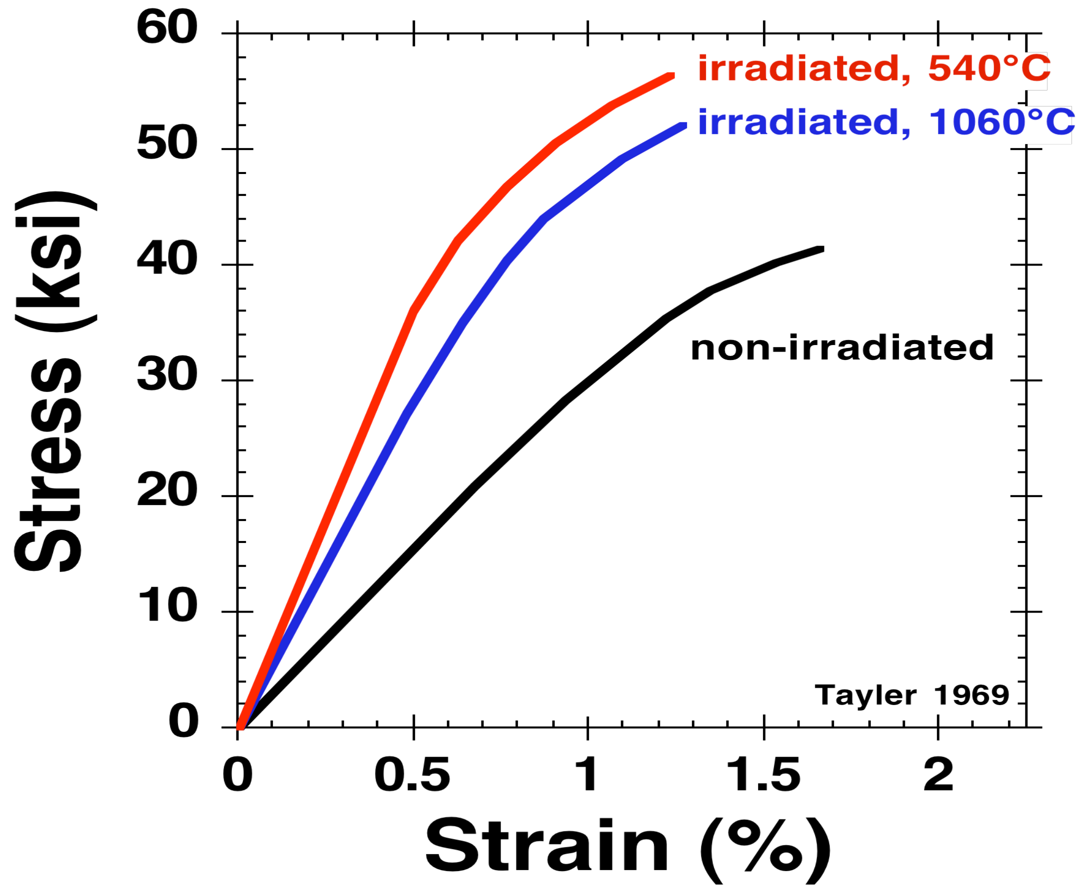
- **Overall Large Volume Increase**
- **<u>Does not</u> conserve volume**

Dimensional Change in Graphite Crystal
High Temperature



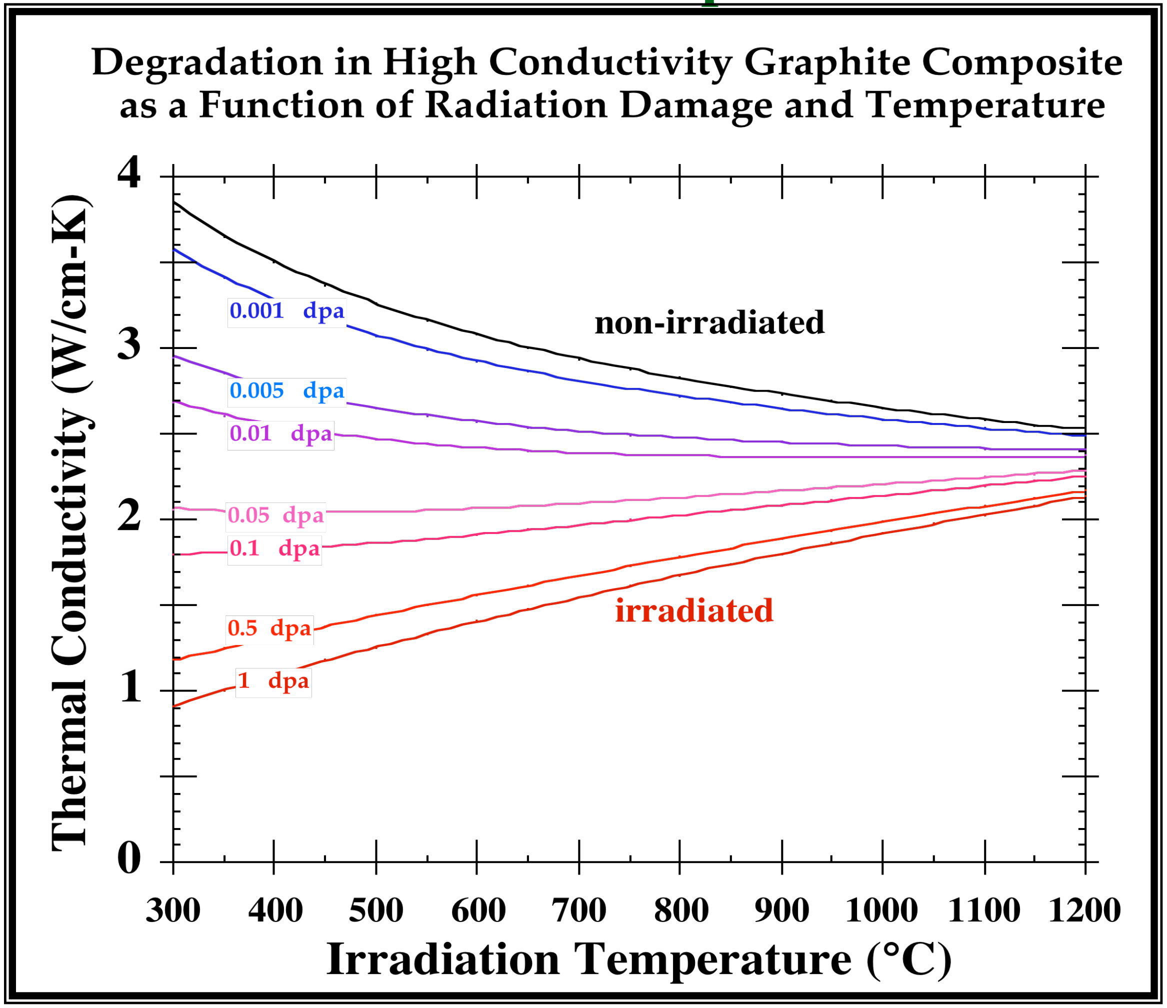
- Above ~1000 K both vacancy and interstitials are mobile.
- Dimensional change occurs to high dose conserving volume.
- Higher crystal perfection material suffers less dimensional change due to enhanced defect recombination.

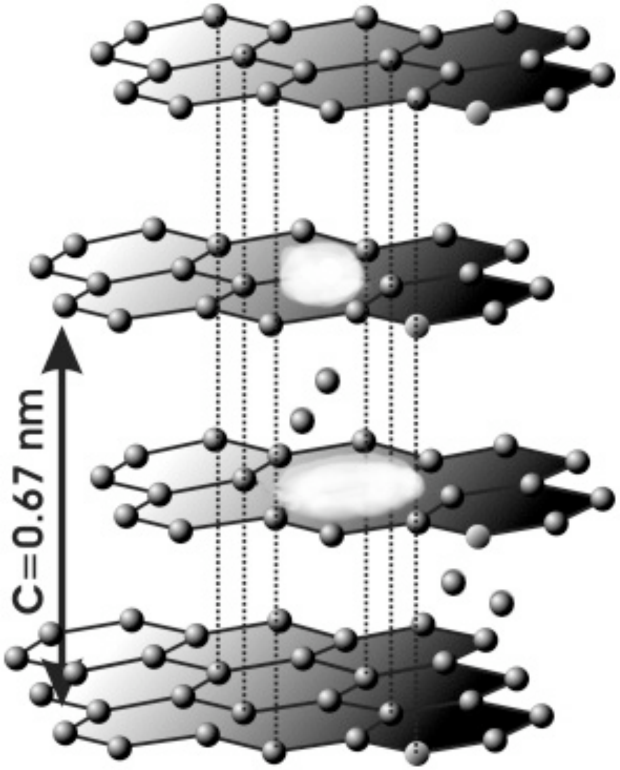
Pyrolytic Graphite : Property Changes Under Irradiation



- Radiation induced defects produce “pinning centers” for migrating defects. These defects are responsible for mechanical properties:
  - resulting in higher elastic modulus upon irradiation.
  - resulting in higher strength upon irradiation.

Irradiation-Degraded Thermal Conductivity
- Graphite -

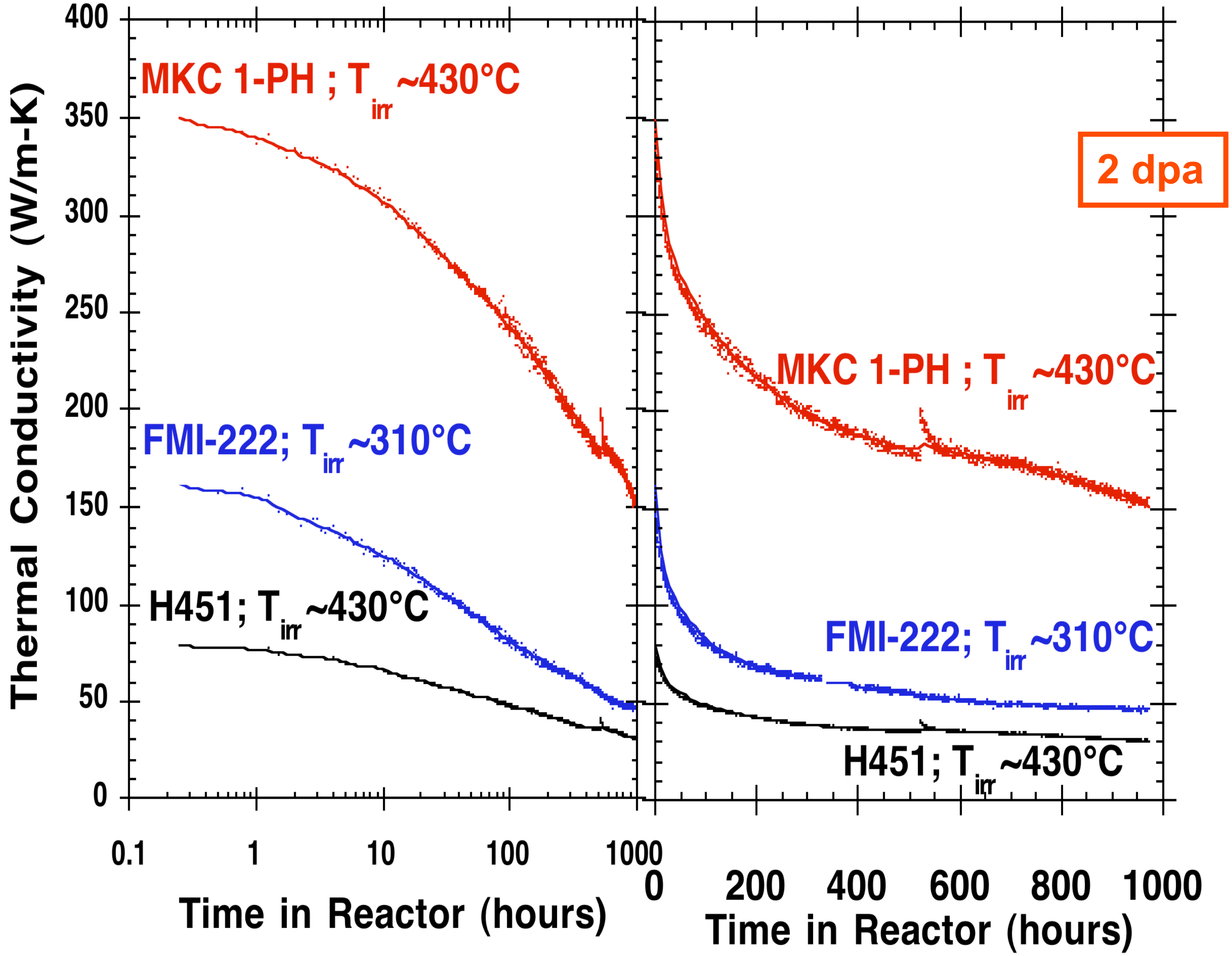




Conductivity reduction ascribed to formation of 4±2 vacancy clusters

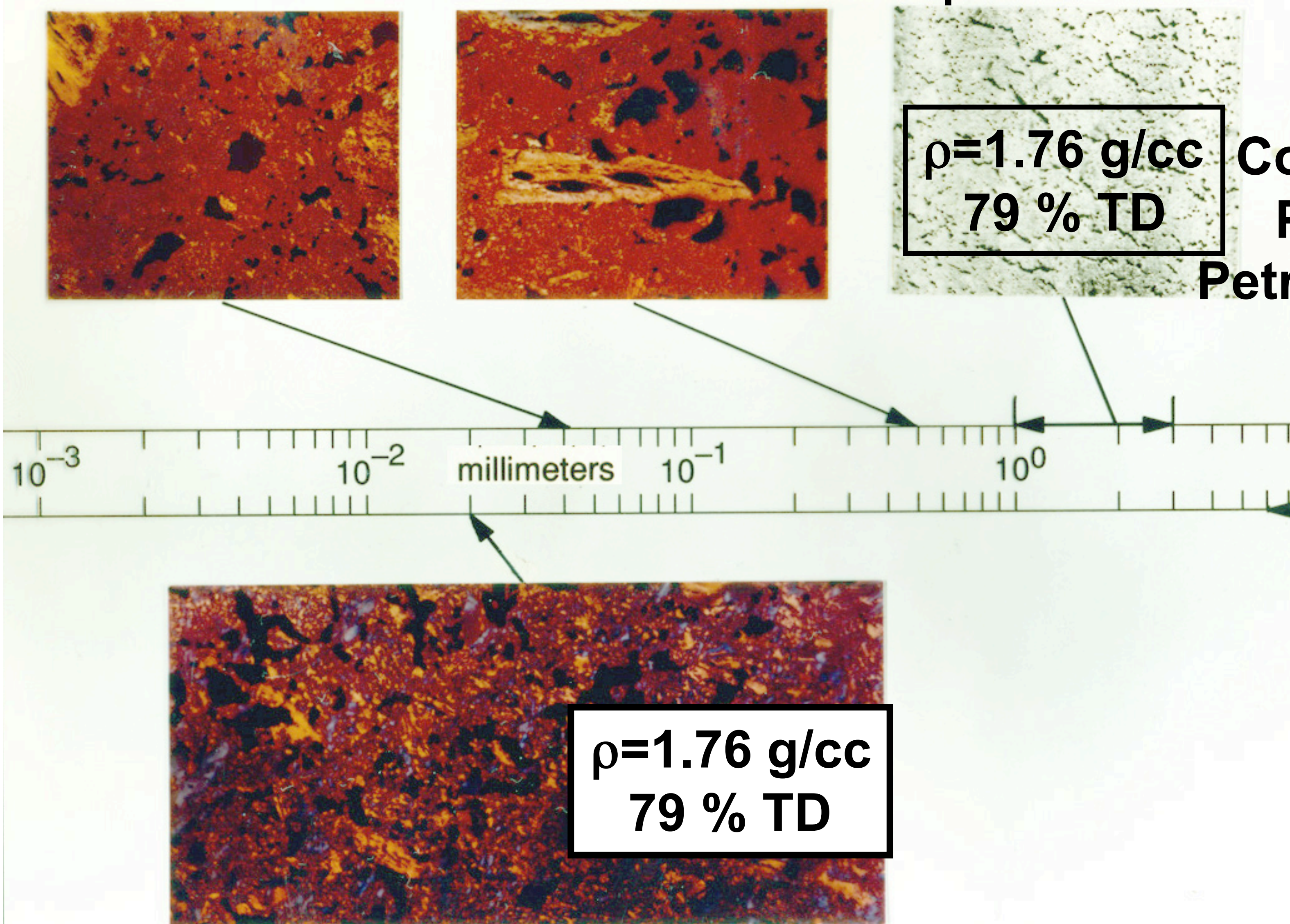
- **Thermal conductivity in graphite is dominated by phonon transport.**
- **Vacancy complexes formed during irradiation are extremely effective at scattering phonons and degrading thermal conductivity.**

Comparison of Thermal Conductivity Degradation



Nuclear Graphite

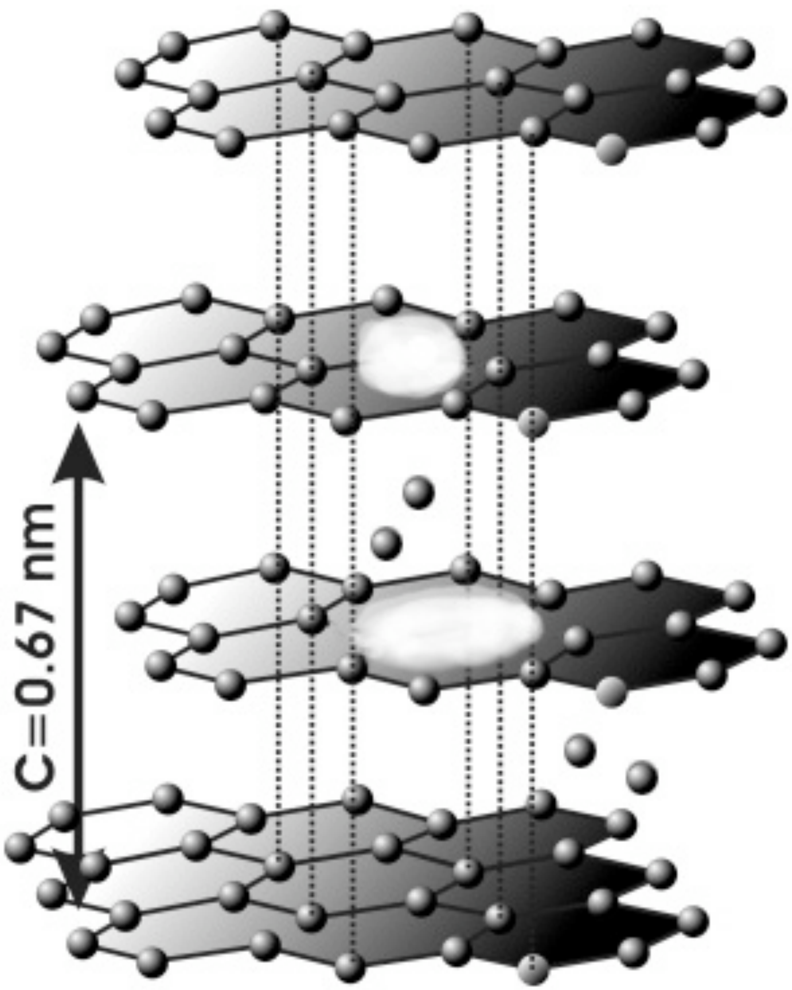
H-451 Extruded Nuclear Grade Graphite

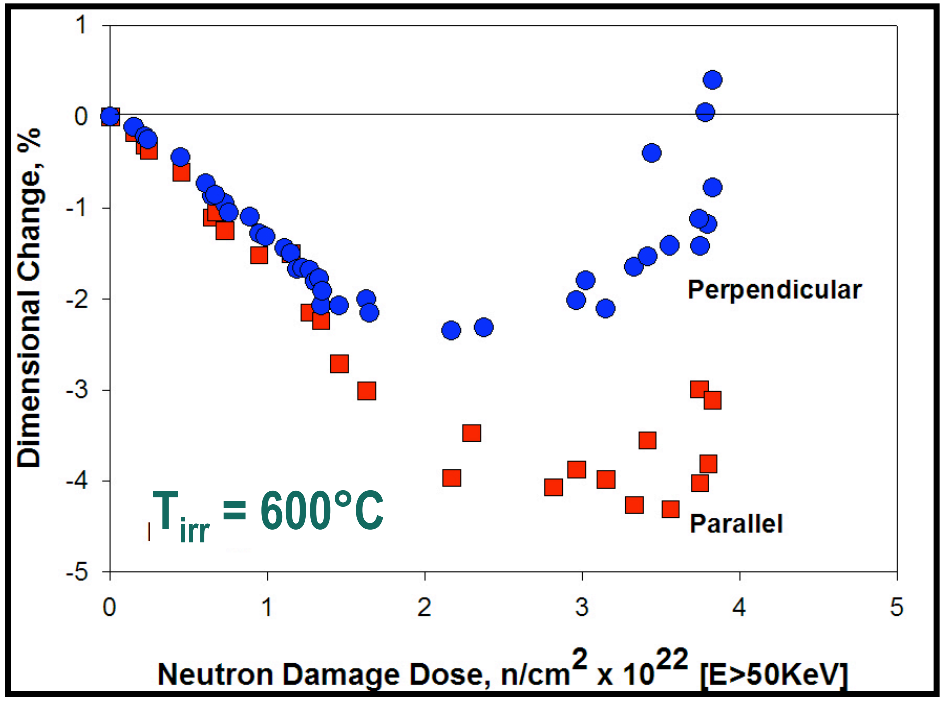


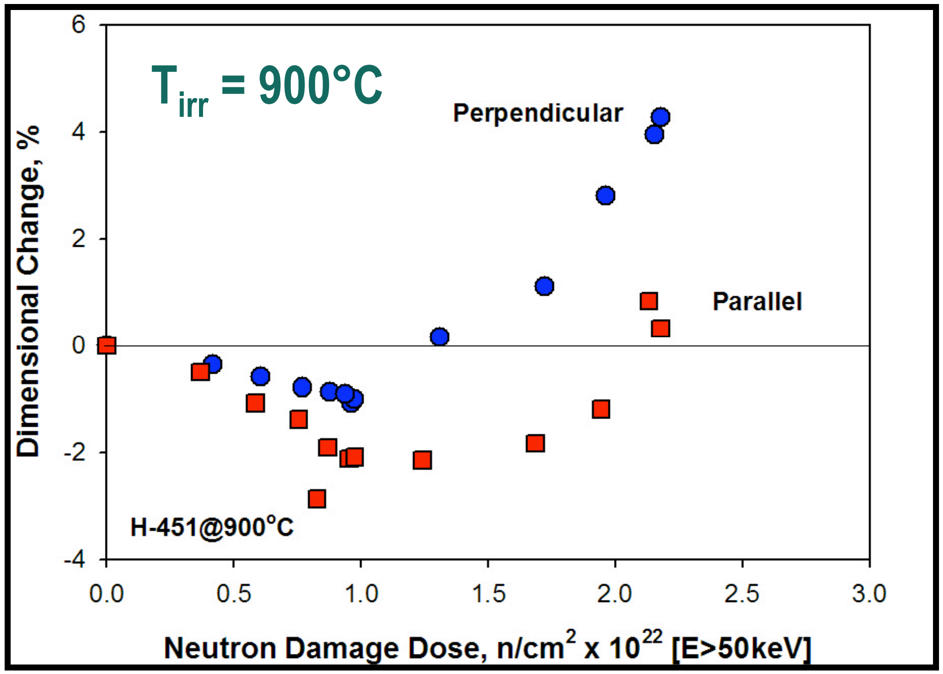
IG-11 Isomolded Nuclear Grade Graphite

<u>Graphite</u>
Coal-tar pitch
Pitch coke
Petrolium binder

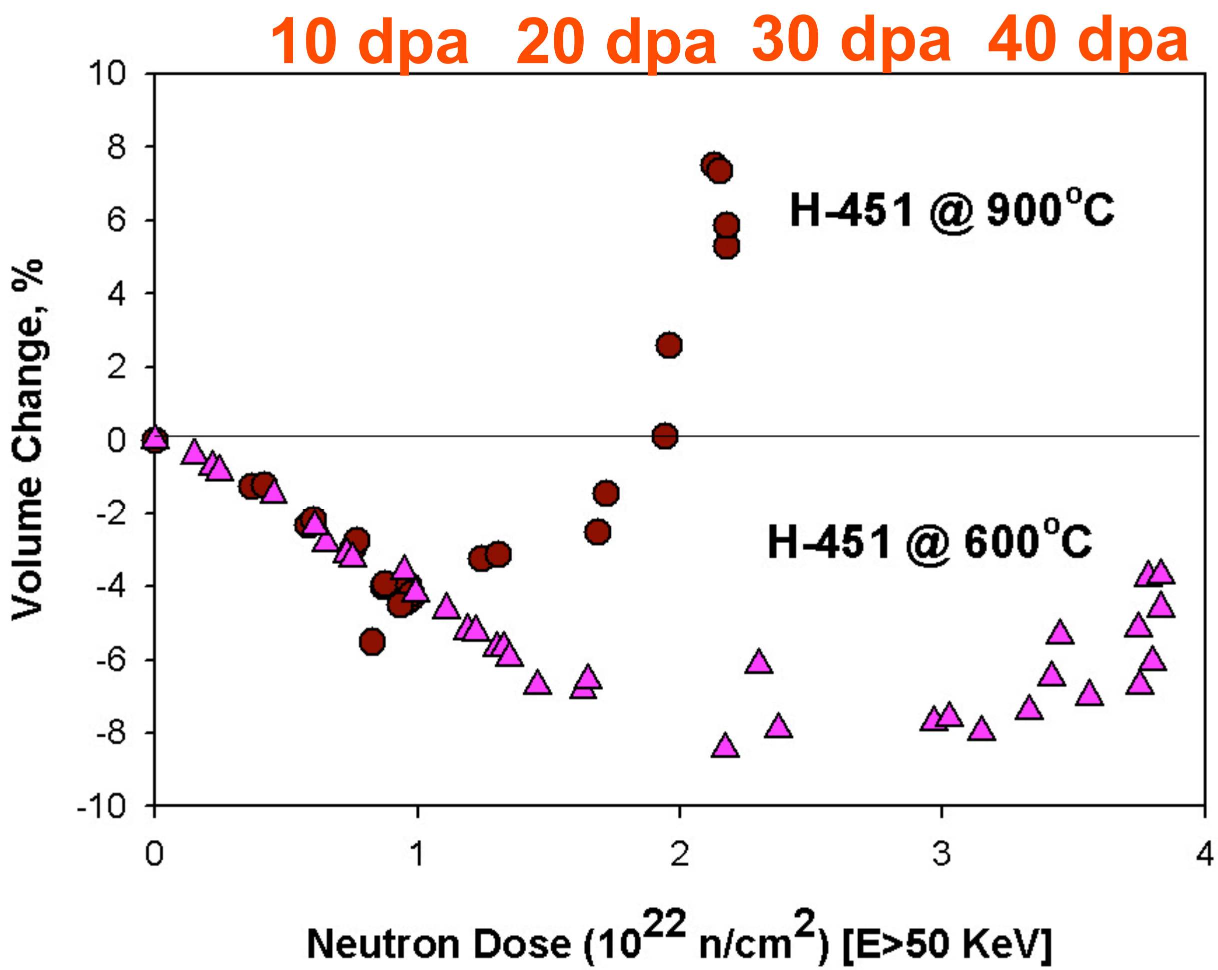
H-451 Nuclear Graphite





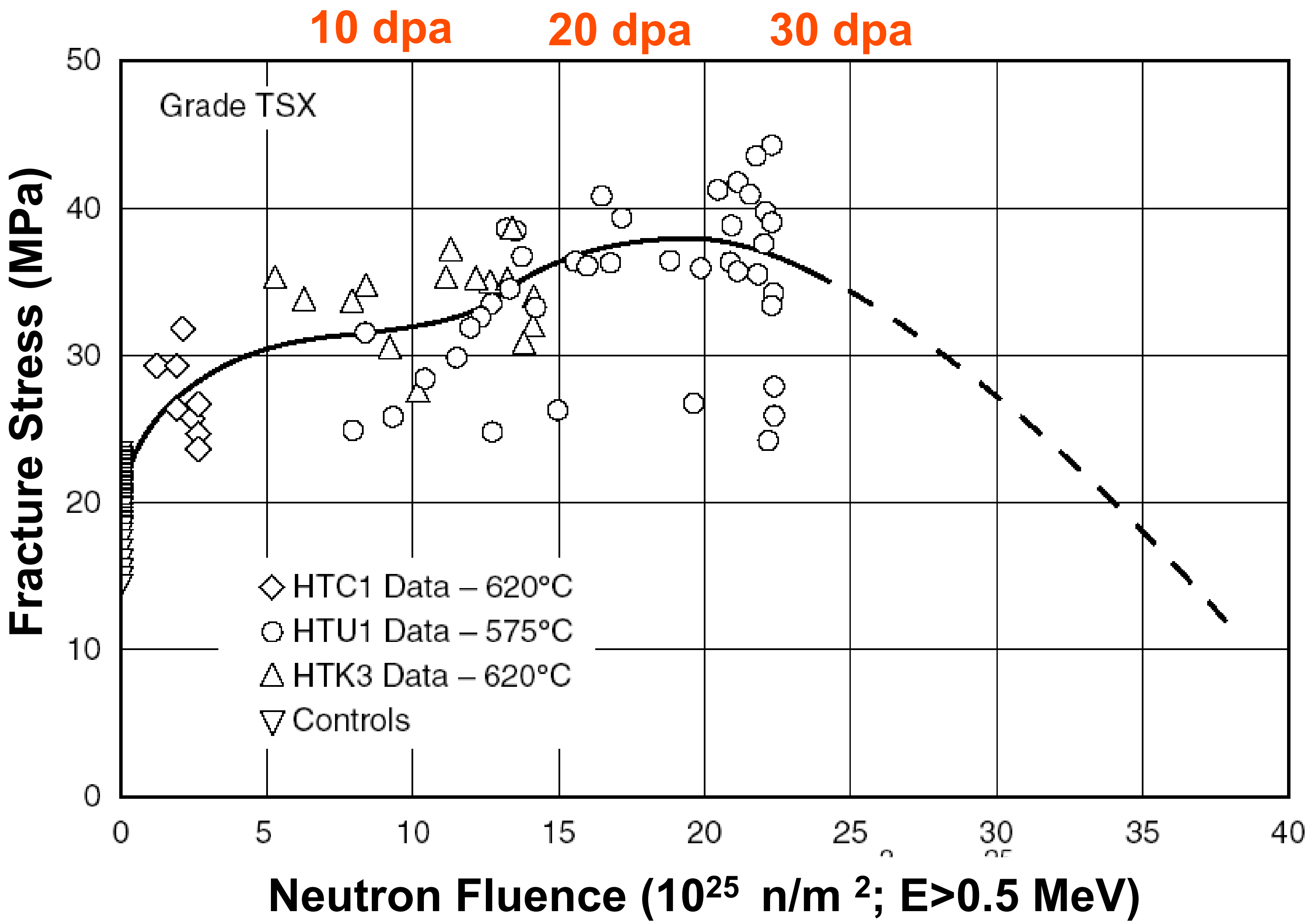


Effect of Temperature and Swelling of Nuclear Graphite

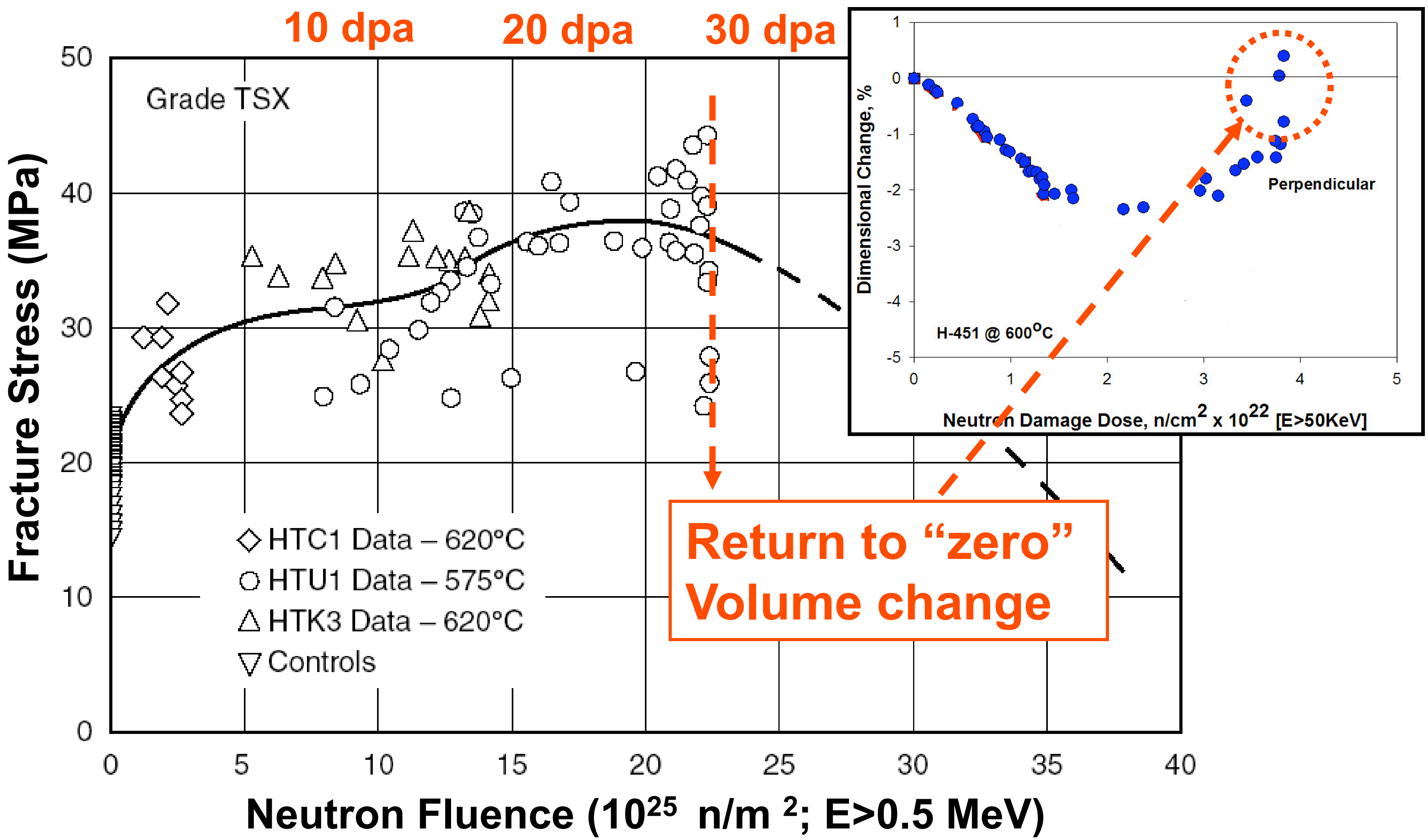


- initial <c> swelling accommodated by closure of intrinsic porosity.
- once porosity filled swelling can begin.
- less initial porosity for higher initial temperature (closure of intrinsic porosity.)

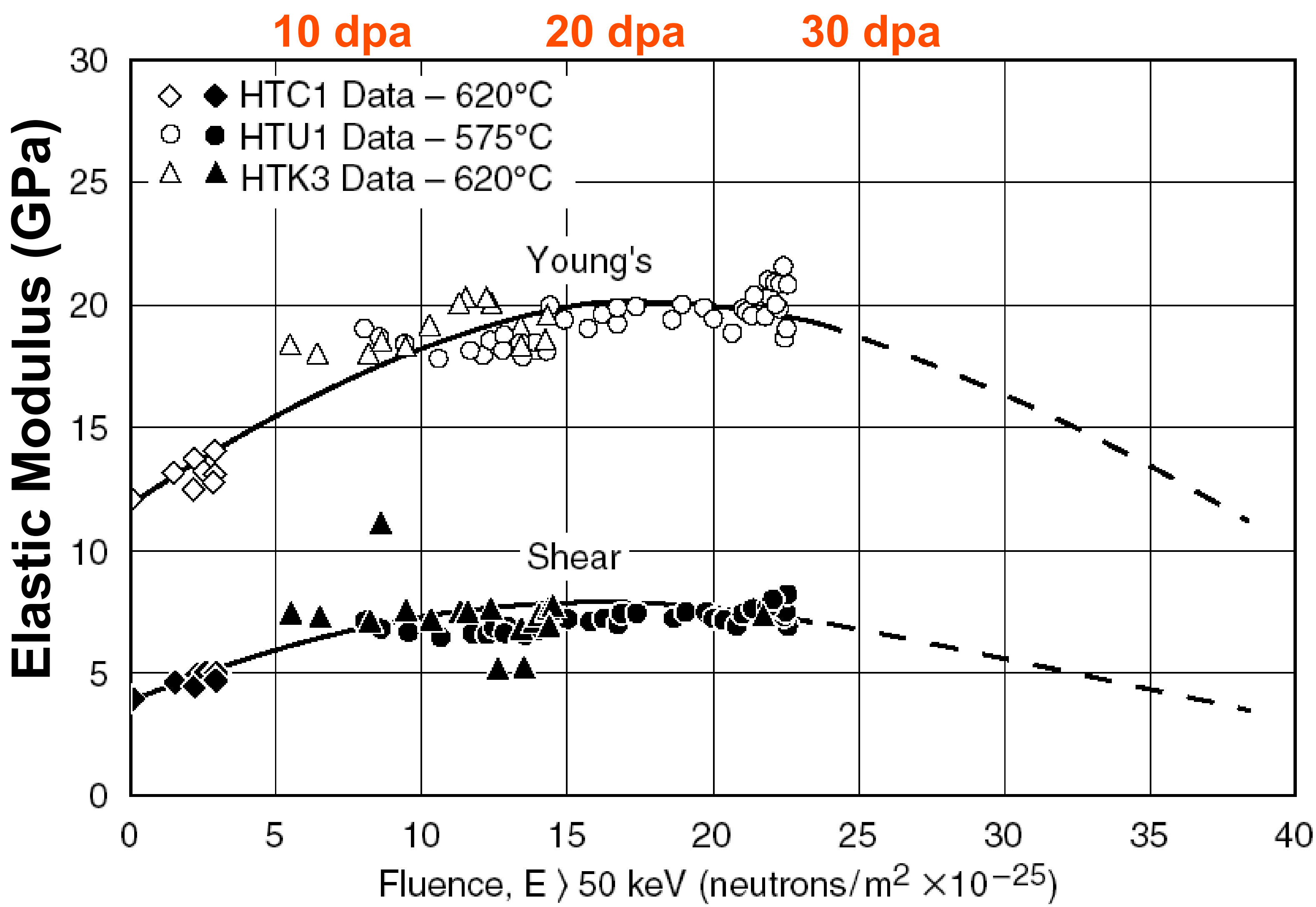
Effect of Irradiation on Strength of Nuclear Graphite



Effect of Irradiation on Strength of Nuclear Graphite



Effect of Irradiation on Elastic Modulus of Nuclear Graphite



Moderators for Modern Gas Cooled Reactors

- Due to the much greater instability of BeO as compared to graphite, graphite is the current and future moderator of choice.

- Current work in the US and throughout the world is to reproduce "bygone" era graphite and qualify it for use in the newer, perhaps higher temperature reactors.

- While there will will not be a much enhanced performance in these graphites, by increasing our understanding of their behavior (creep an oxidation as example,) greater margin can be gained in their use.

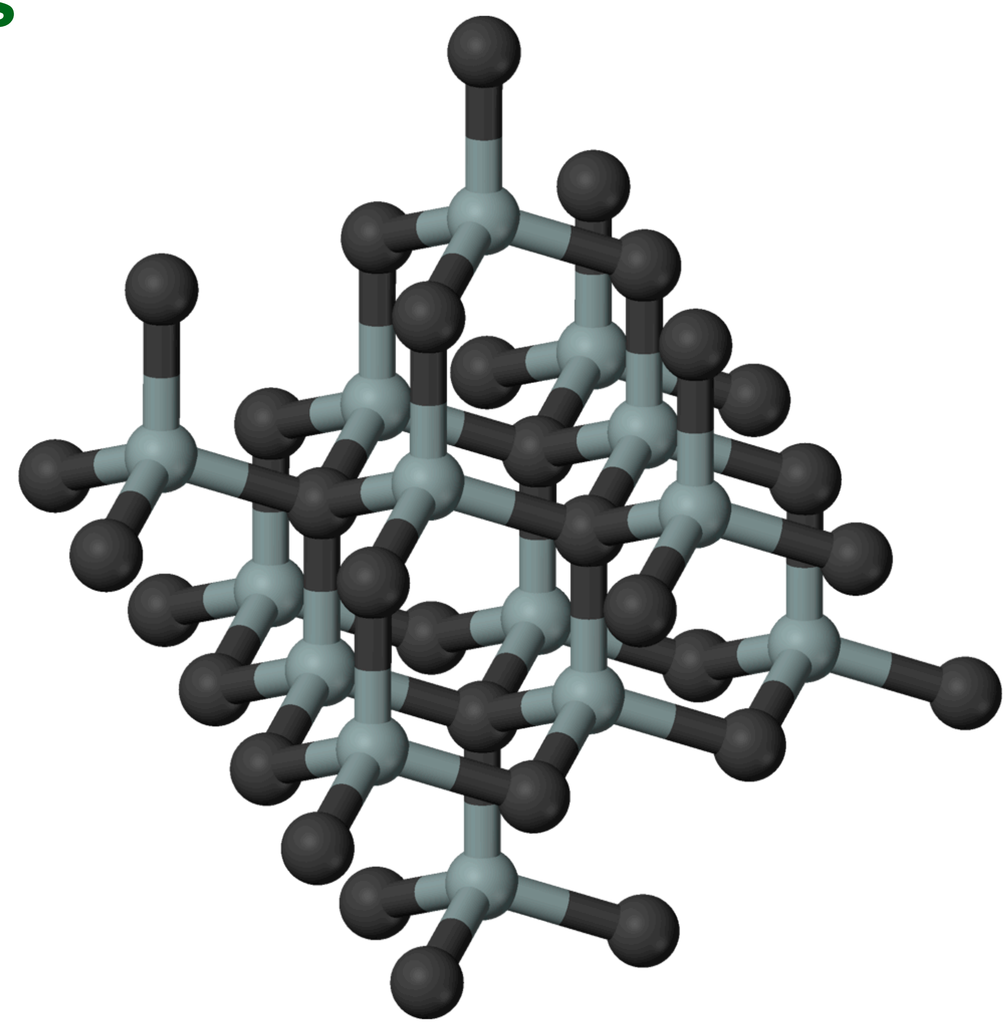
SiC As Clad in HTGR's

• SiC has two common forms:

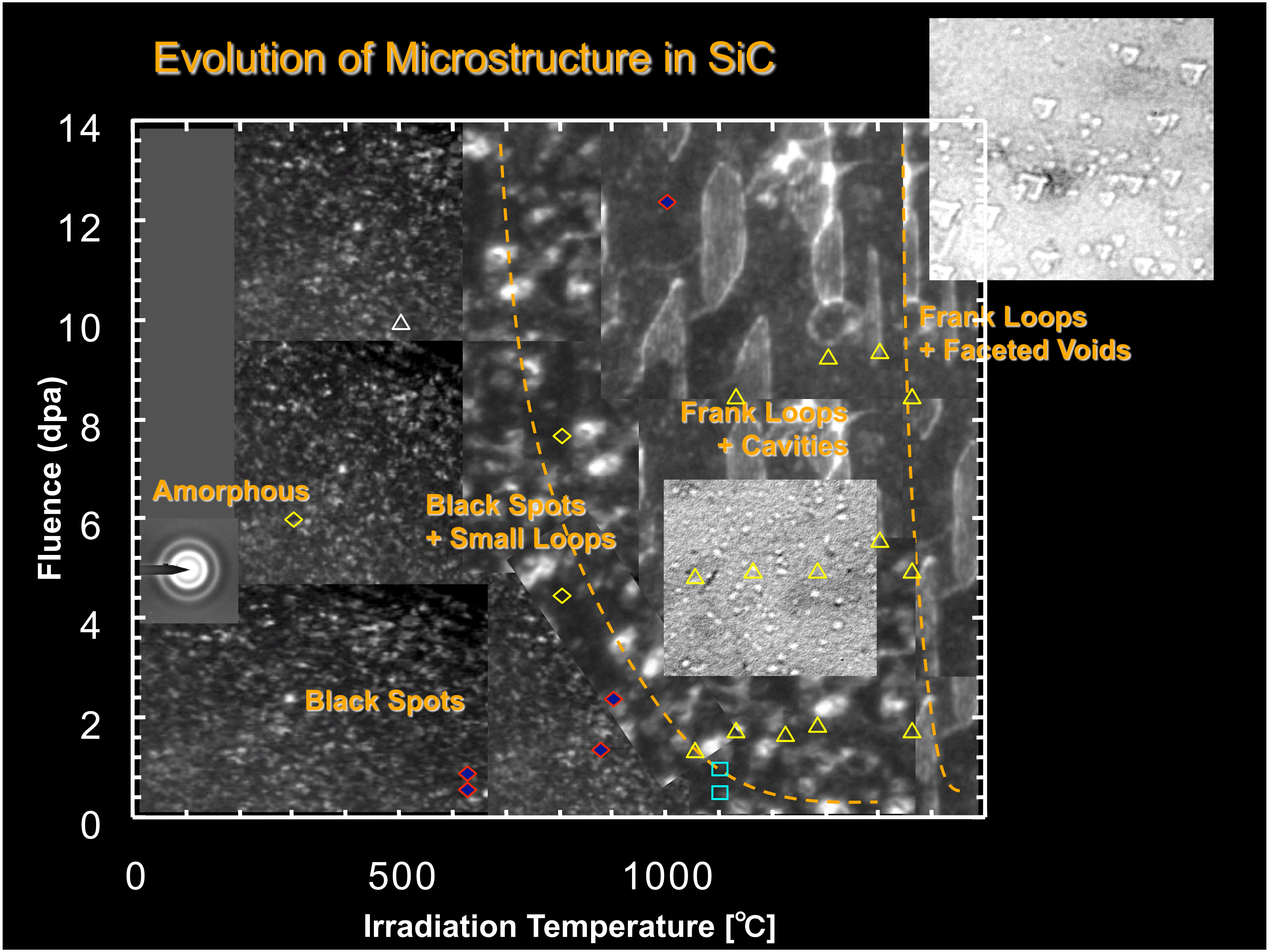
α-SiC : hexagonal
>100 polytypes

β-SiC : FCC

• α-SiC is the most common polytype and the focus of all nuclear application.

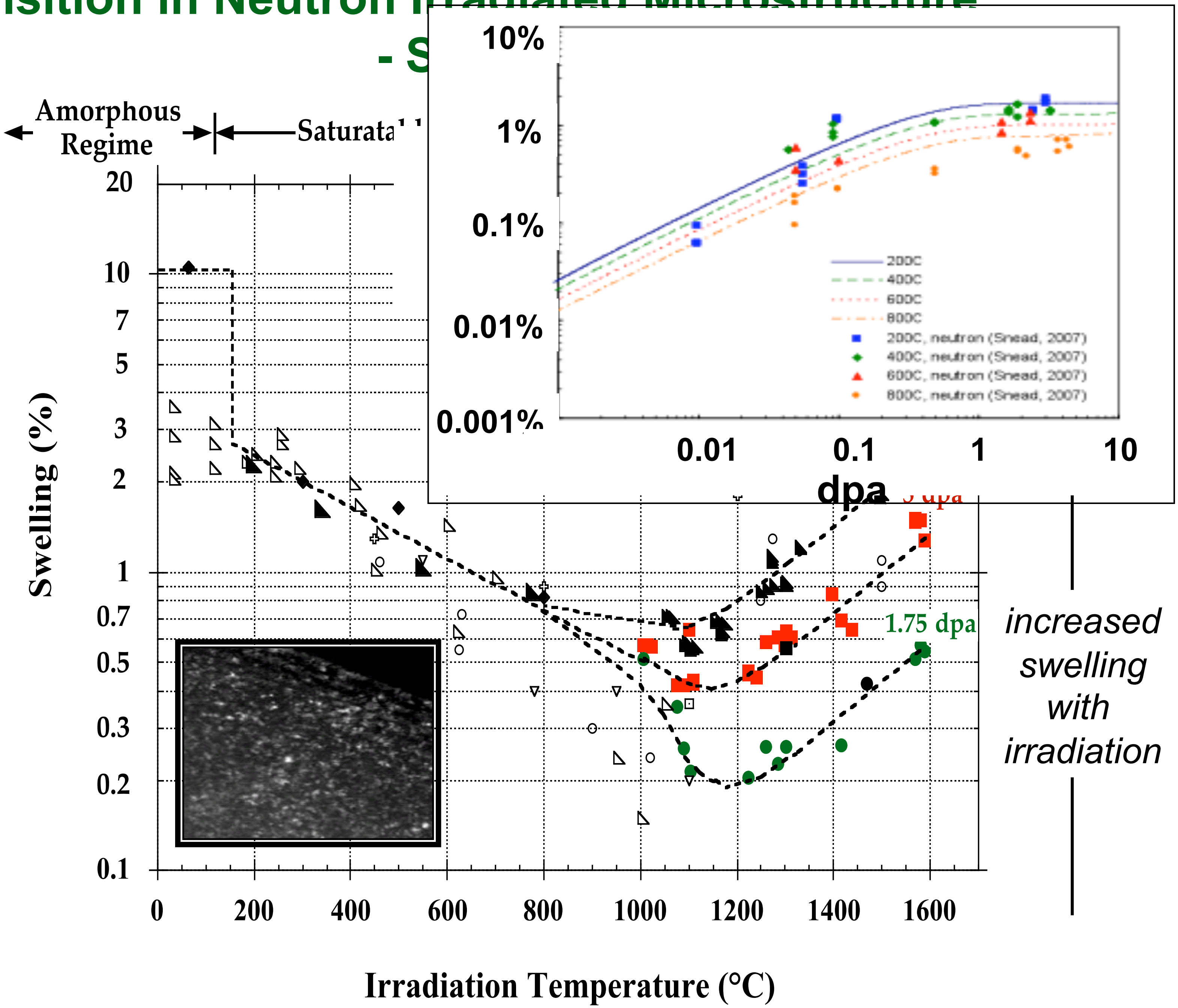


• The reported defect migration kinetics range dramatically though both carbon and silicon interstitials appear to be mobile for T>~400K, vacancies ~ >1200K.



Evolution of Microstructure in SiC
Fluence (dpa)
0
2
4
6
8
10
12
14
Irradiation Temperature [℃]
0
500
1000
Amorphous
Black Spots
Black Spots
+ Small Loops
Frank Loops
+ Cavities
Frank Loops
+ Faceted Voids

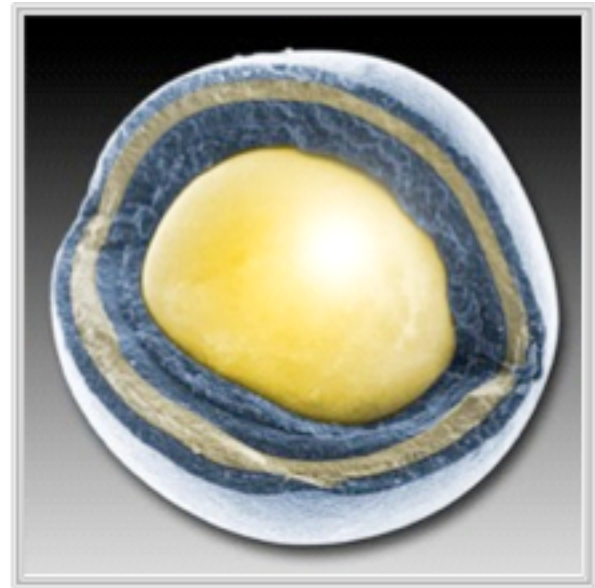
Transition in Neutron Irradiated Microstructure
- S

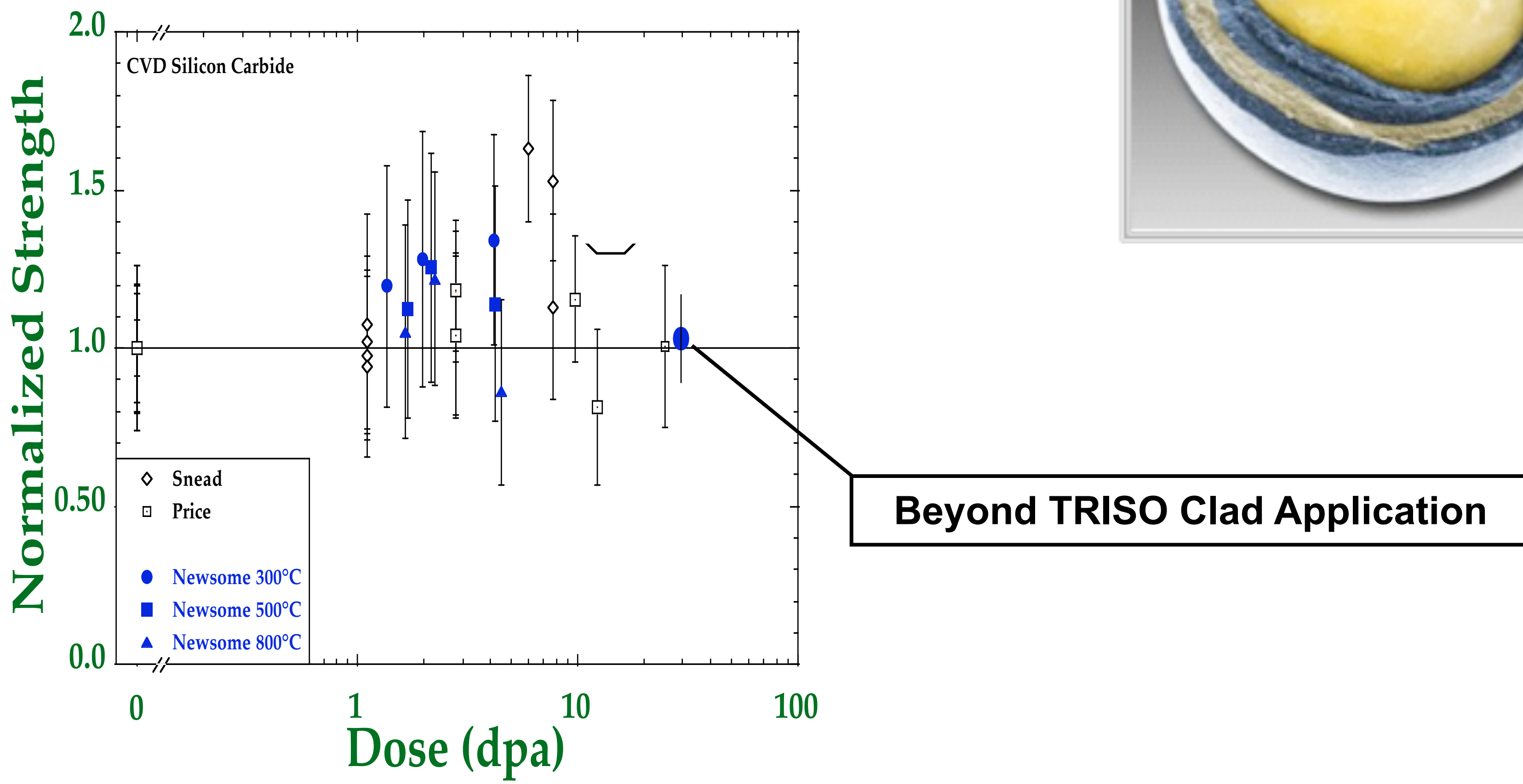


increased swelling with irradiation

Use of TRISO AGR Fuels:

•ZERO fuel failures for ~ 19% fertile fuel burnup



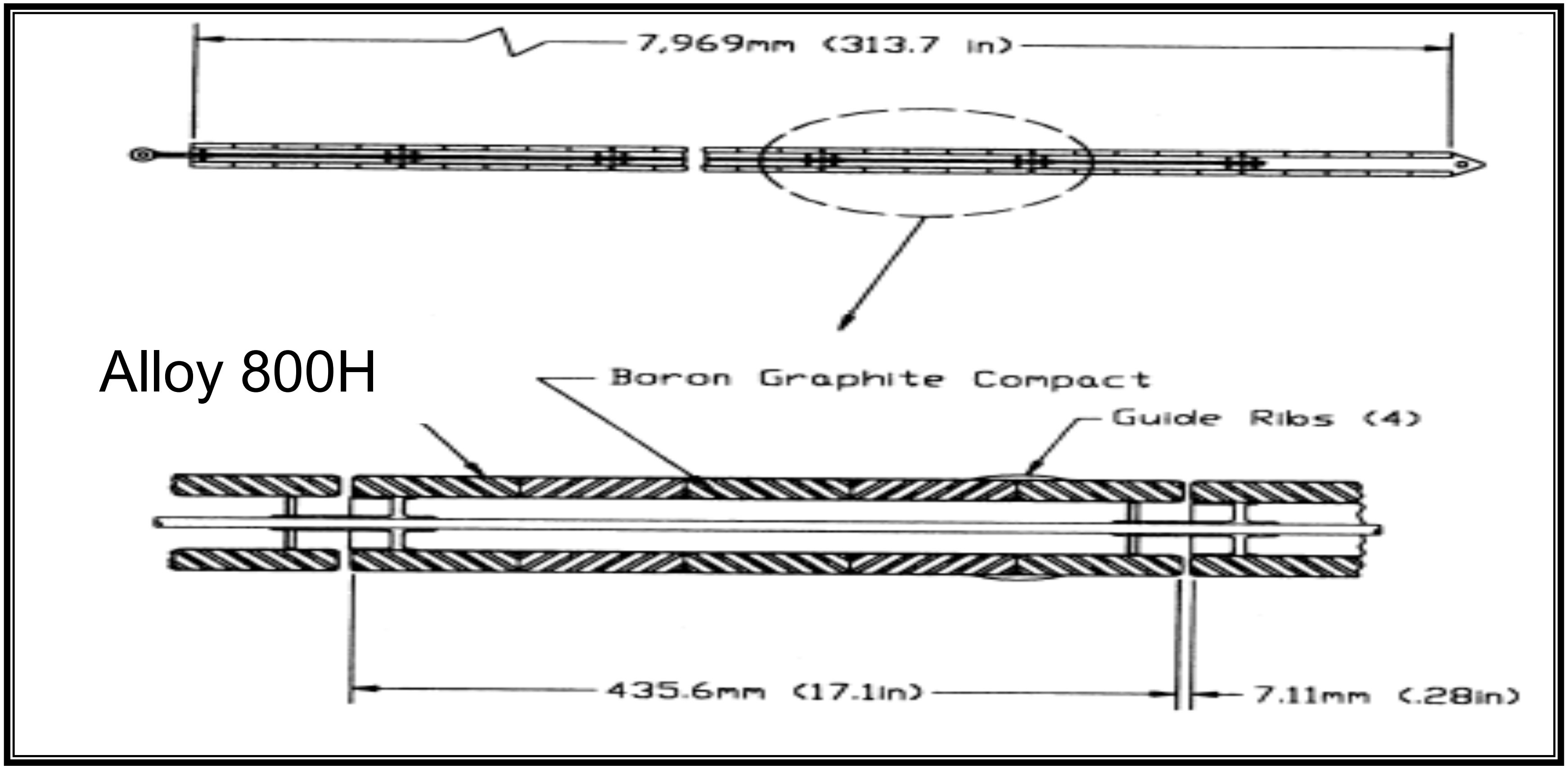


- **Of the crystalline ceramics studied to date, SiC has demonstrated the best retention of strength and dimensional stability to high-dose.**

In Core Structural Materials Beyond AGR's

GT-MHR Control Rod Concept

(Courtesy of General Atomics)



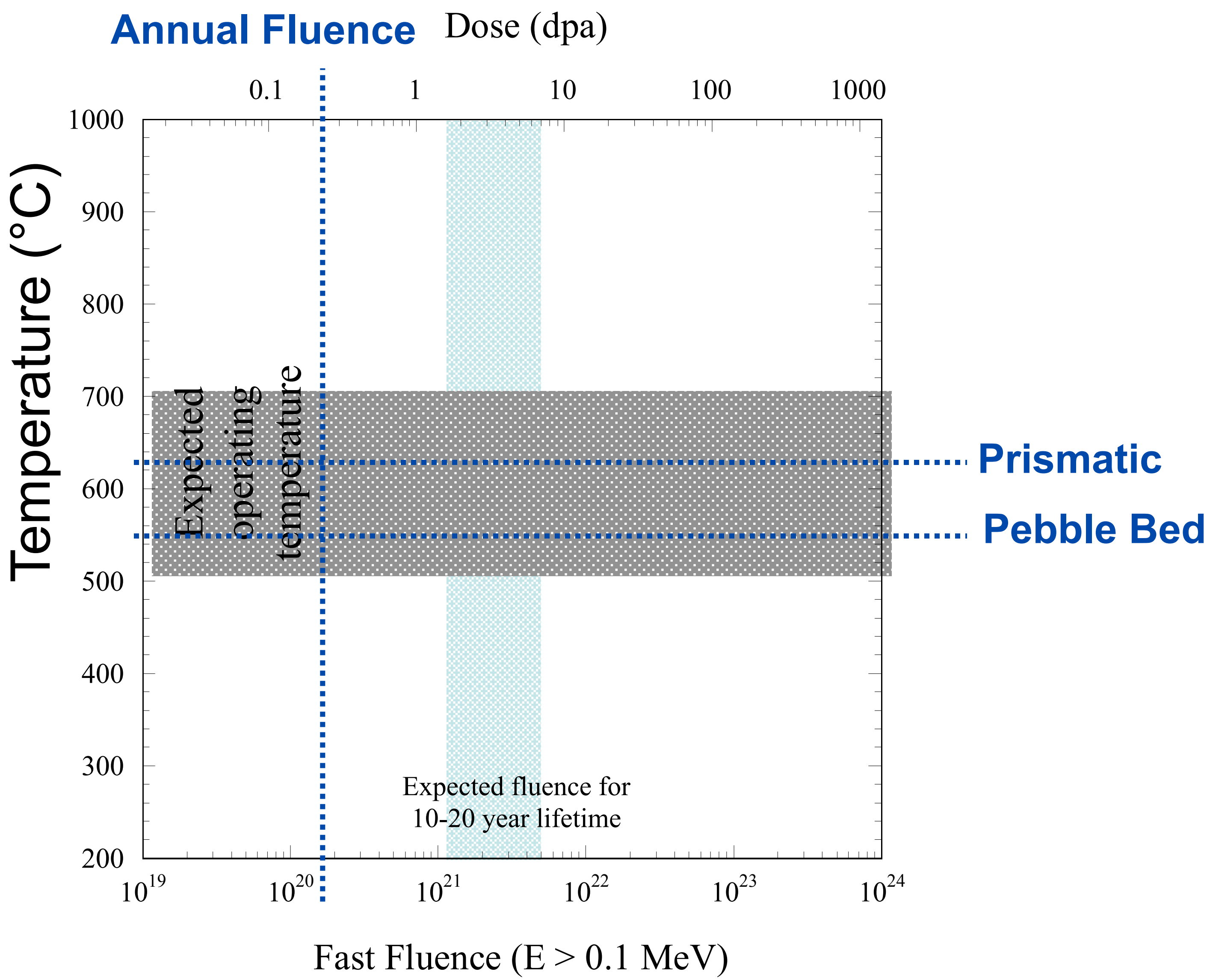
NGNP Operating conditions

- **Incoloy 800H is proposed as a control rod material for NGNP with 950°C outlet temperature.**

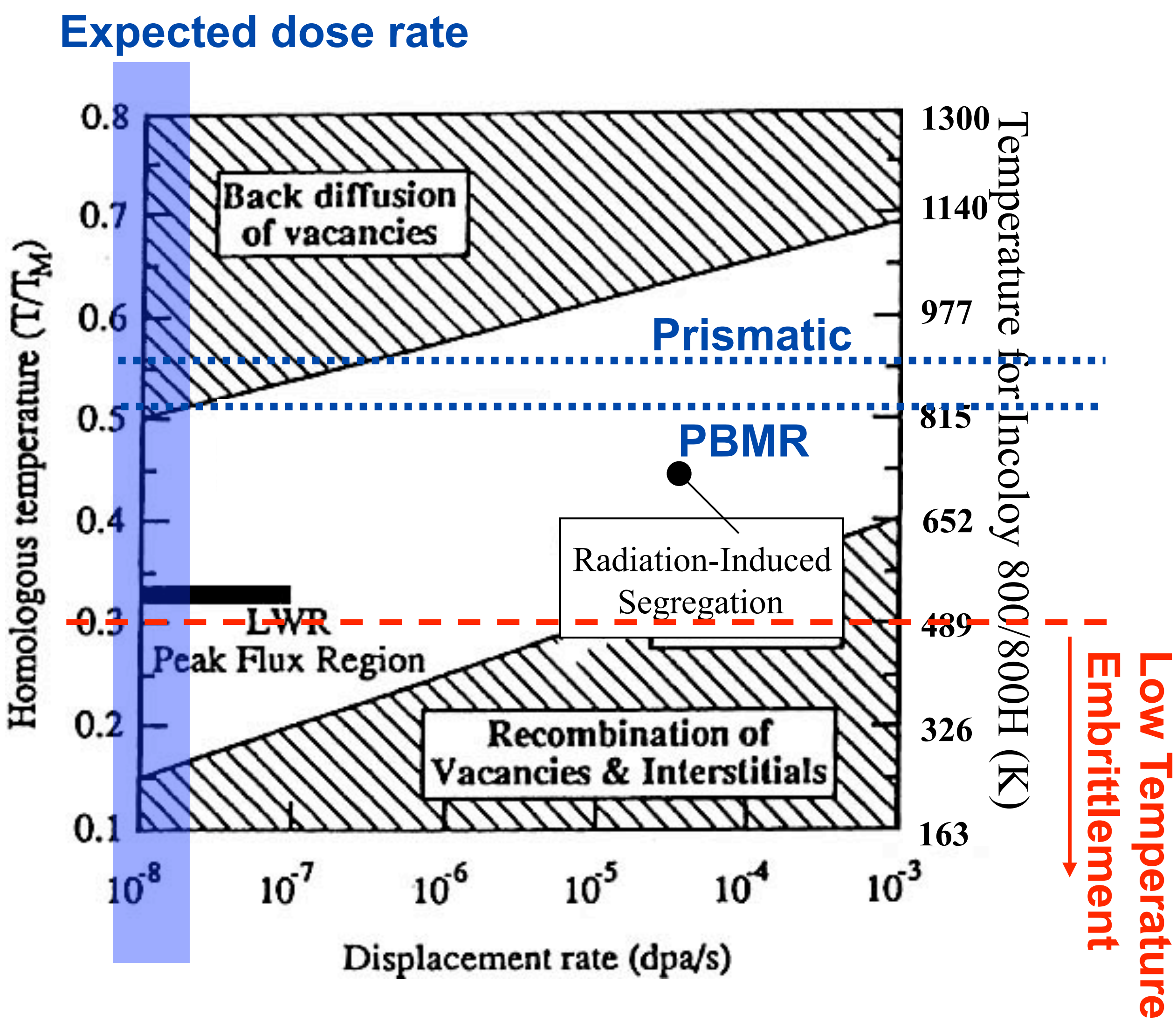
- **Pebble Bed:**
  - **Flux: ~0.2 dpa/year**
  - **Operating temperature: 550°C**
  - **Max excursion temperature: 1000°C**

- **Prismatic:**
  - **Flux: ~0.2 dpa/year**
  - **Operating temperature: 630°C**
  - **Max excursion temperature: 1175°C**

NGNP Operating Regime



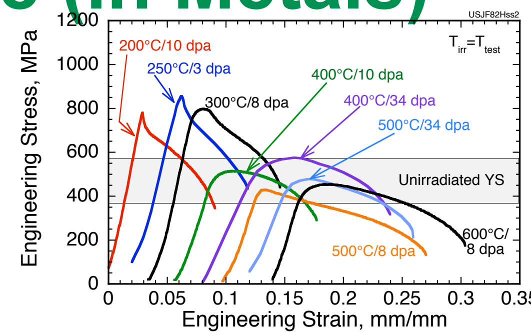
Expected Irradiation Degradation for Alloy 800H



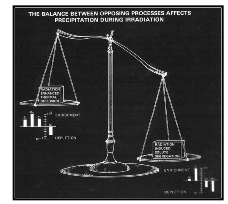
- Operating temperatures are well above where low-temperature embrittlement is expected. Just above irradiation creep?
- Radiation-induced segregation will not likely play a dominant role.
- Radiation-enhanced diffusion may be important as it could help drive changes in precipitate structure.
- Swelling and He generation (and embrittlement?) are likely important.

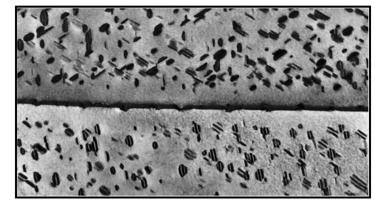
Five Evils of Radiation Damage (in Metals)

- **Radiation hardening & embrittlement ($<0.4\ T_M$, >0.1 dpa)**



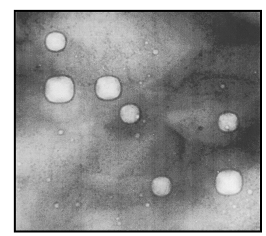
- **Phase instabilities from radiation-induced precipitation (0.3-0.6 T_M, >10 dpa)**



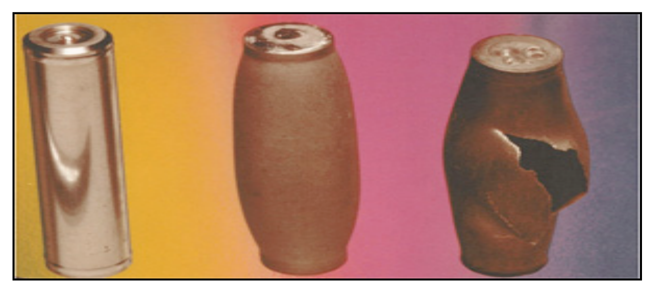


- **High temperature He embrittlement ($>0.5\ T_M$, >10 dpa)**

- **Volumetric swelling from void formation (0.3-0.6 T_M, >10 dpa)**



- **Irradiation creep ($<0.45\ T_M$, >10 dpa)**



He generation and accumulation

- **He content is likely an important factor in high temperature irradiation resistance of Incoloy 800.**

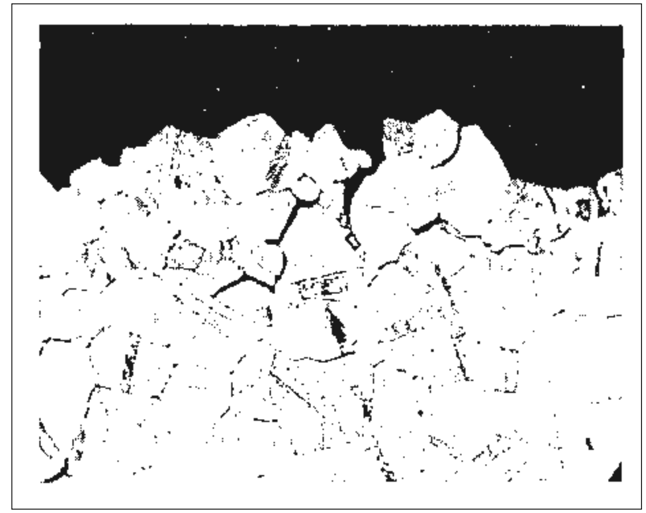
- **He will be generated in control rod material and Incoloy itself**
  - $^{10}B\ (n,\alpha) \rightarrow {}^{7}Li$
  - $^{58}Ni\ (n,\gamma) \rightarrow {}^{59}Ni\ (n,\alpha) \rightarrow {}^{56}Fe$

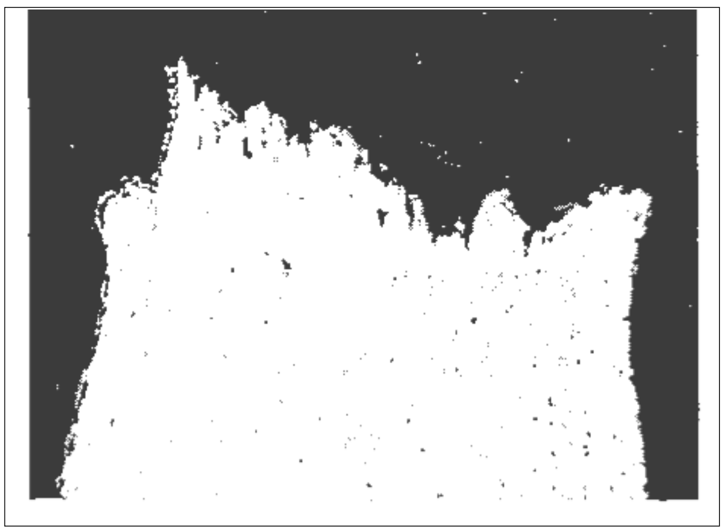
- **High He concentrations could lead to helium grain boundary embrittlement which has been observed in other austenitic alloys.**

High Temperature Grain Boundary He Embrittlement

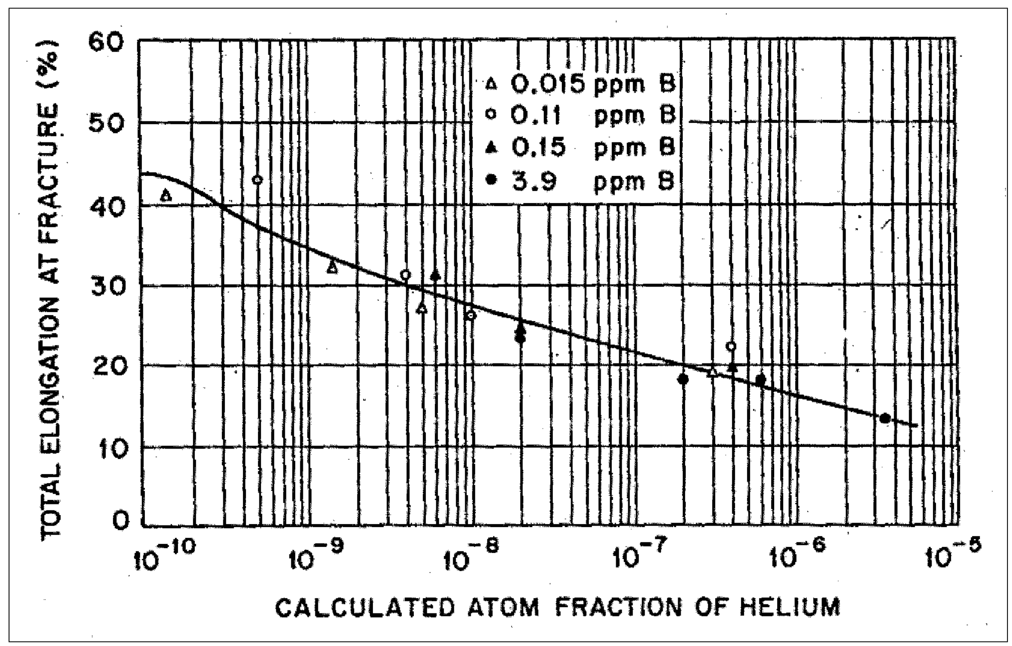
- **The conventional view is that He-embrittlement is a high temperature (> T_m/2) slow strain rate phenomenon caused by stress-induced growth of He bubbles at grain boundaries.**

IG failure in irradiated 316 SS

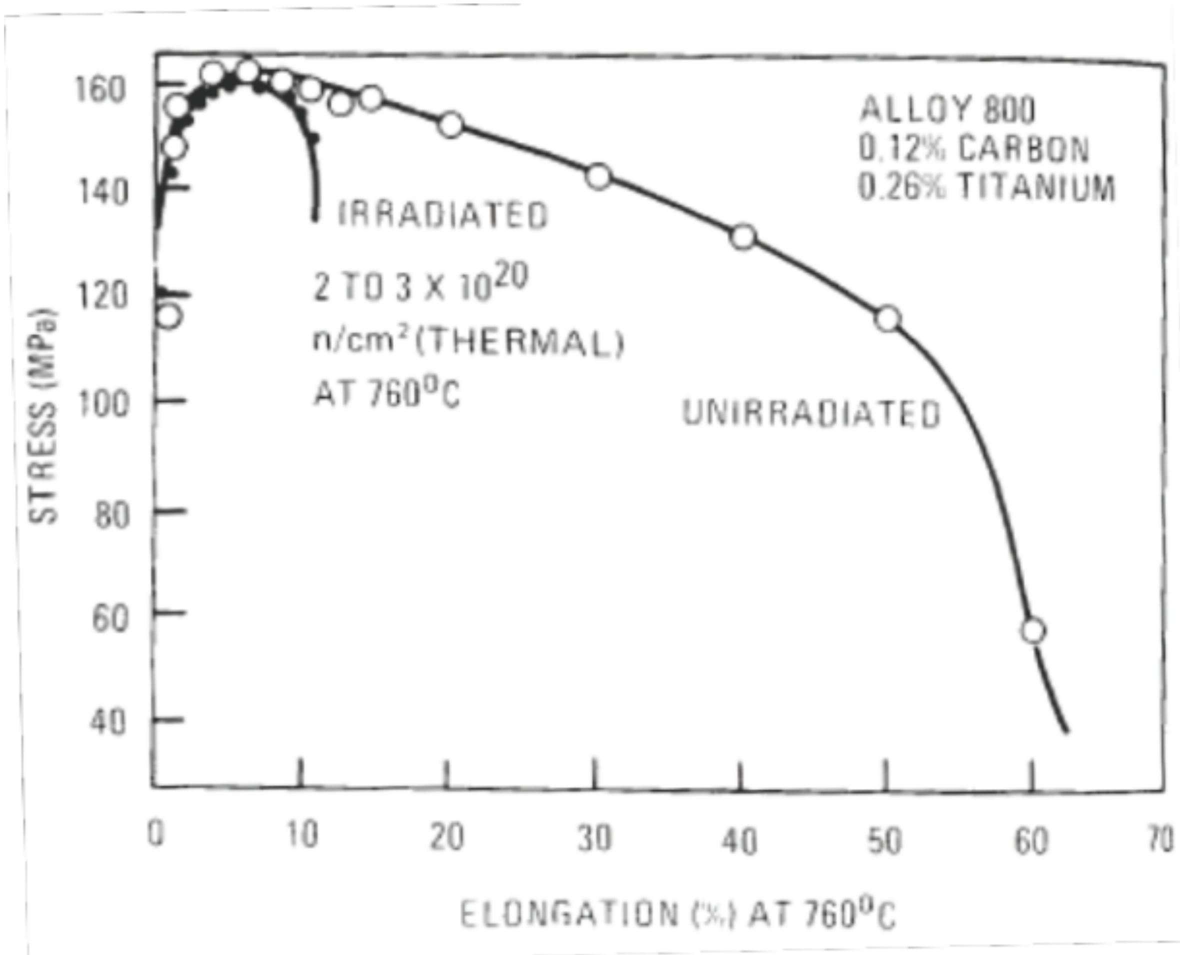




Ductile failure in non-irradiated 316 SS



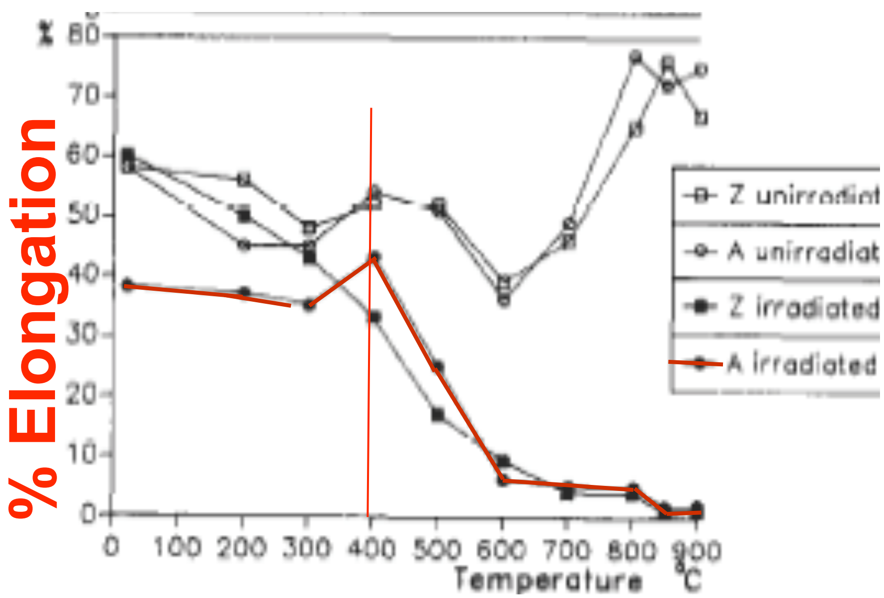
Harman Data



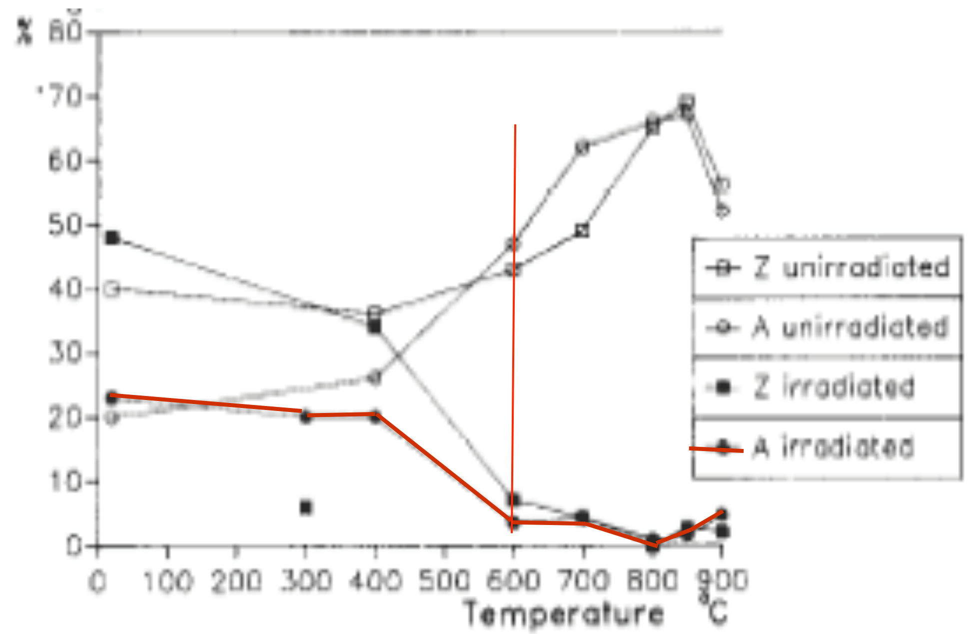
- Harman tested Incoloy 800 after irradiation at 760°C to ~0.1 dpa.
- Irradiated elongation dropped to under 10%.
- No appreciable change in YS.

Source: [4] D.G. Harman, Post-irradiation tensile and creep rupture properties of several experimental heats of Incoloy 800 at 700 and 760 C," ORNL Report TM-2305, 1968.

Incoloy 800, T_{irr} = 400°C



Incoloy 800, T_{irr} = 600°C

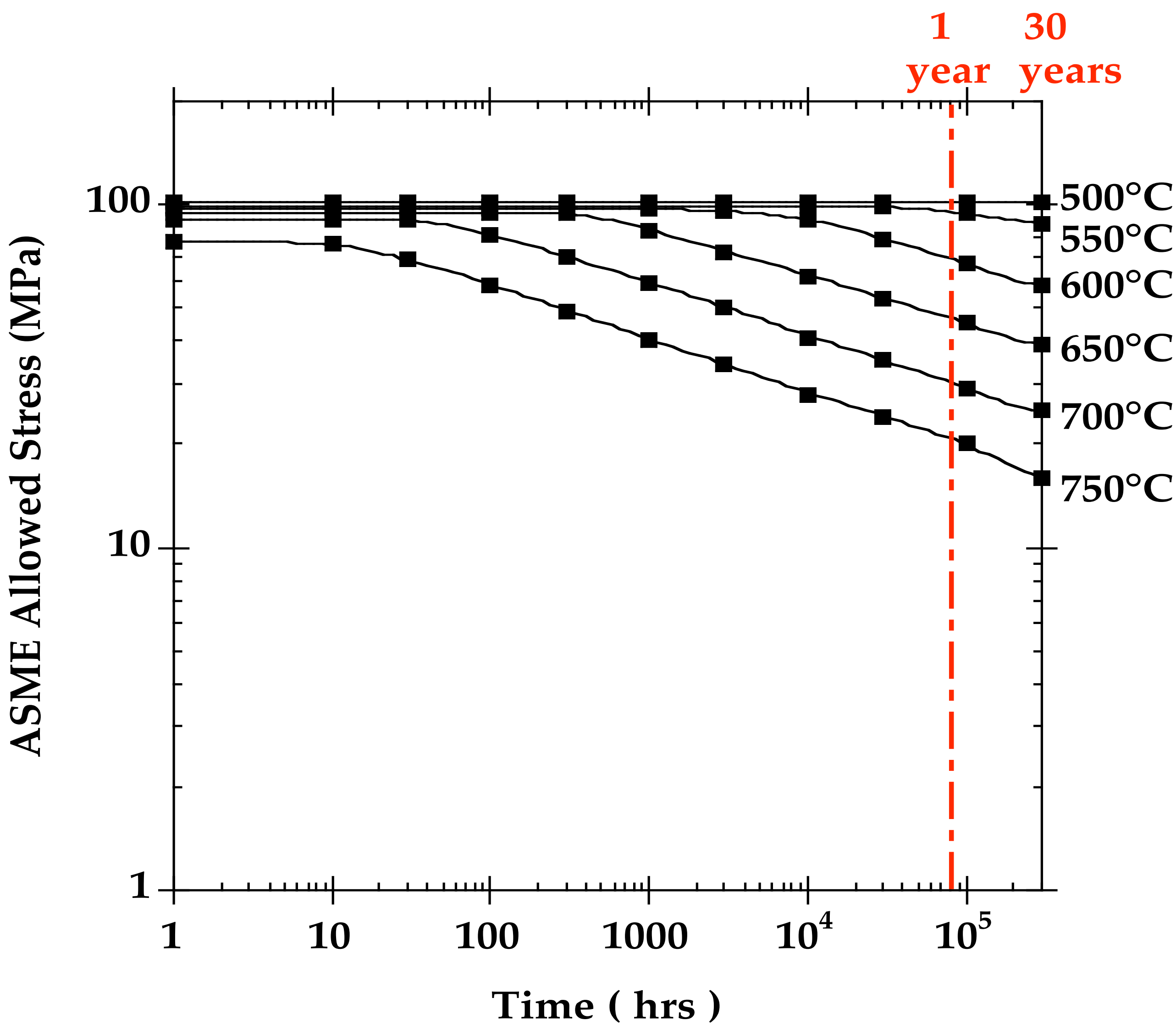


Z = fracture elongation, A = total elongation

- **Thiele reports elongations of Alloy 800 after $1x10^{21}$ n/cm^2, various temperatures**
- **Failure elongation is below 5% after irradiation and testing at 600°C.**
- **Above irradiation temperature, ductility significantly drops.**
- **Recent data of Nanstad reports similar behavior.**

Source: [7] Thiele et al., “Influence of Test Temperature on Post-irradiation, High temperature tensile and creep properties of X8CrNiMoNb 1616, X10NiCrAlTi 32 20 (Alloy 800) and NiCr22Fe18Mo (Hastelloy X),” J. of Nucl. Mater., 171 (1990) p. 94-102.

Alloy 800H ASME Stress Allowables : Non-irradiated



Harman Data

Alloy 800
T_{irr}=760°C
~ 0.1 dpa

Creep Rupture

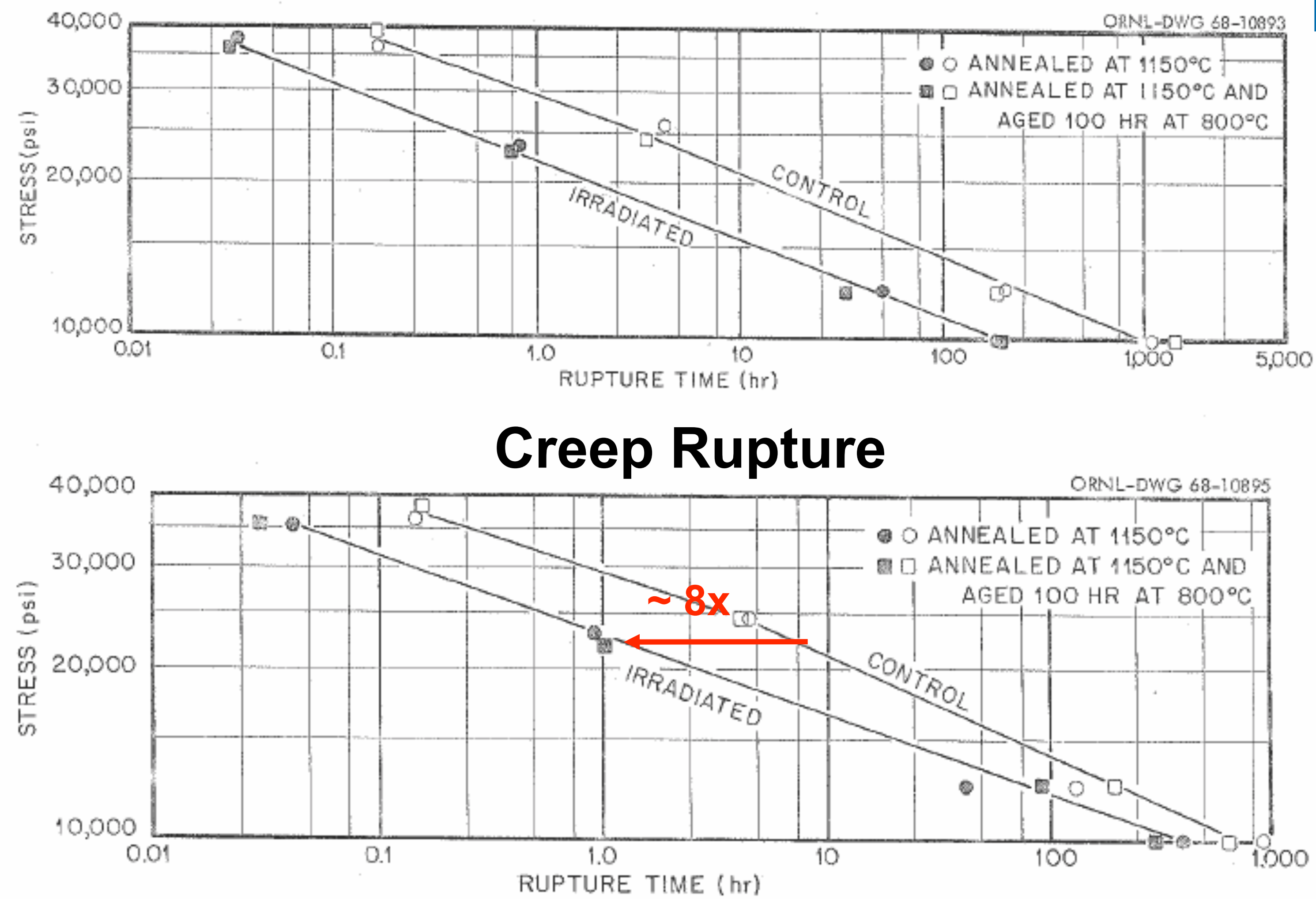
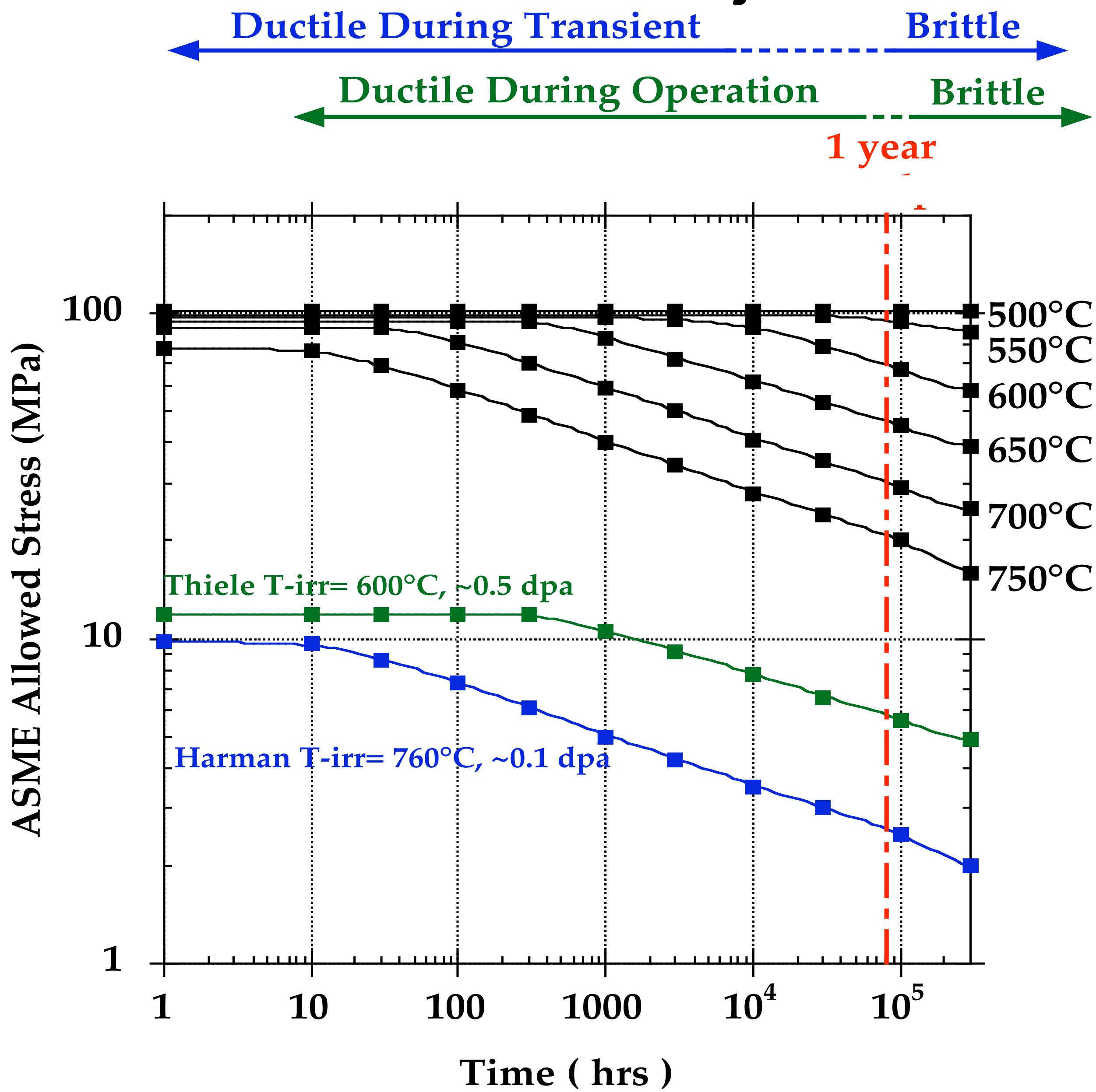


Fig. 14. Creep Rupture of Experimental Incoloy 800 Containing 0.13% C and (a) 0.17% or (b) 0.10% Ti. Specimens were irradiated or soaked (controls) at 760°C and tested in air at 760°C. Both tensile and creep data are plotted.

- Harman(0.1 dpa), Thiele(~1 dpa), and Watanabe(~0.2 dpa) find similar reductions in creep rupture life for Alloy 800.

Source: [4] D.G. Harman, Post-irradiation tensile and creep rupture properties of several experimental heats of Incoloy 800 at 700 and 760 C," ORNL Report TM-2305, 1968.

Alloy 800H Stress Allowables
ASME Non-irradiated and Projected Irradiated



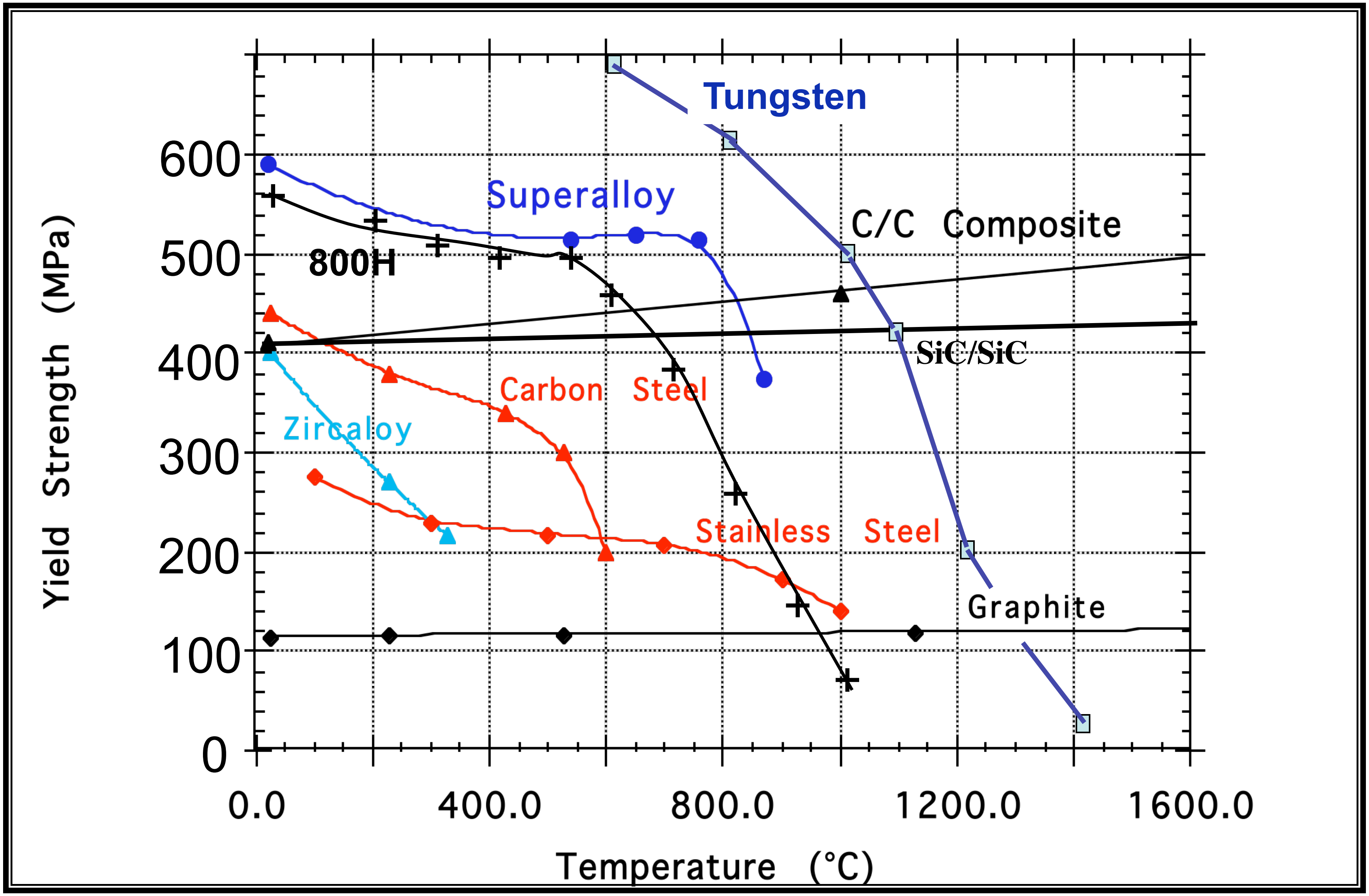
Summary Issues for 800H for Control Rod Application

- Alloy 800H has a distinct advantage over contending Gen IV control rod materials as it is covered under ASME Section III, Subsection NH currently to 760°C <u>and likely later to 900°C.</u>

- There is little data on the effect of irradiation on Alloy 800**H** and the existing data suggests severe embrittlement at HTGR relevant temperature and doses. Such behavior is consistent with high nickel alloys

- Design with Alloy 800H will require use of material with very limited creep rupture strength and ductility.

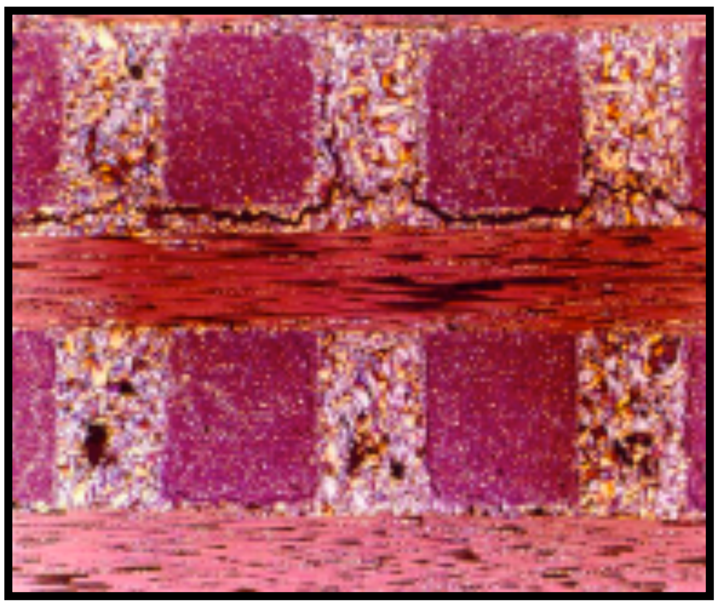
Yield Strength of Various Structural Materials

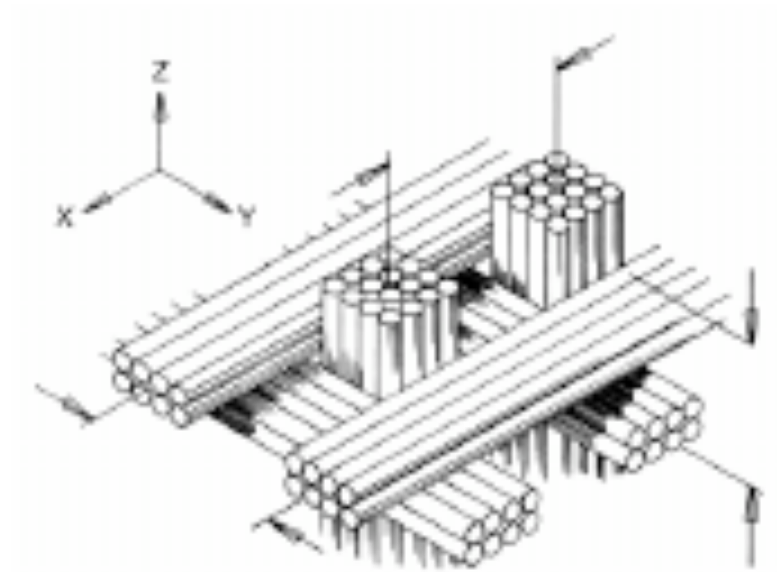


Ceramic Composite Materials

• Composites as being defined here are technically “continuous fiber reinforced composites,” the two most mature of which are Carbon(graphite) Fiber Composites (CFC’s) and Silicon Carbide fiber composites (SiC/SiC.)

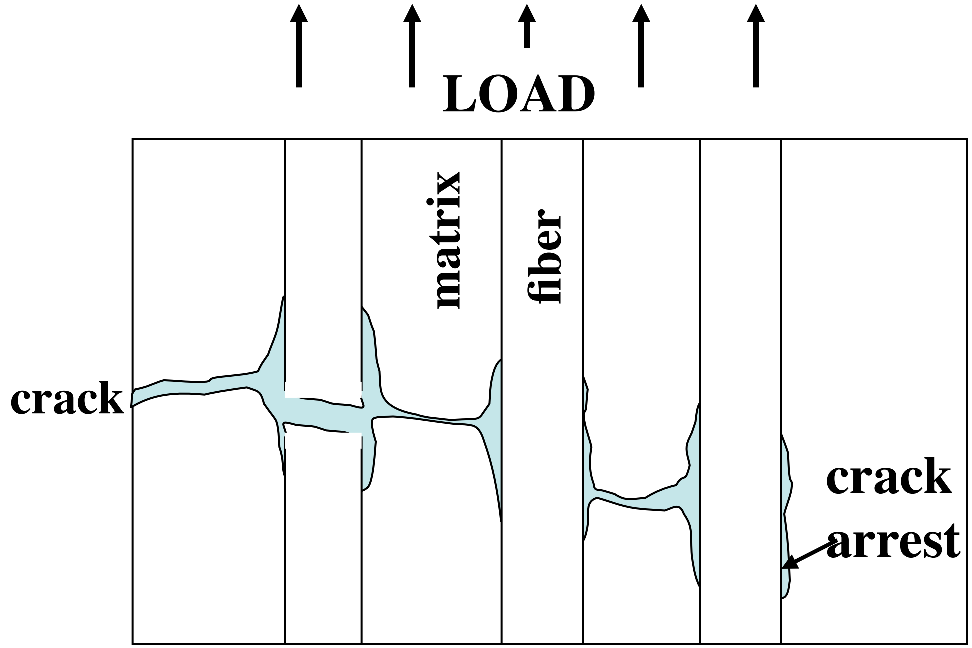
• Of the two, the CFC is the more mature system, though they are similar in terms of processing status and cost.





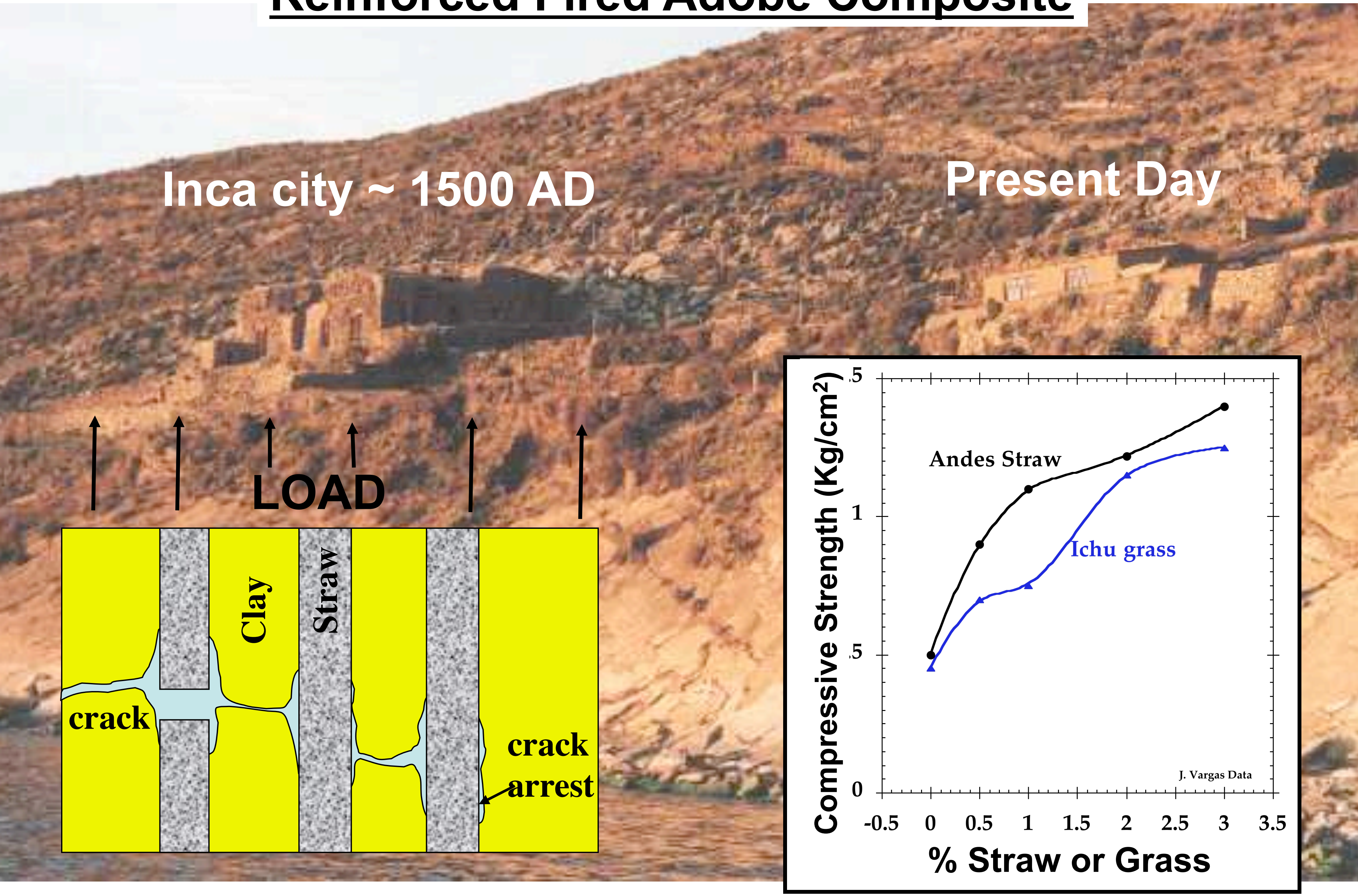
Composite -vs- Monolithic Ceramics

	Toughness $MPa/m^{1/2}$
Steel	> 50
Tungsten	< 20
Monolithic Ceramic	3
Platelet Reinforced Ceramic	6
Chopped Fiber Reinforced	10
Continuous Fiber Reinforced Ceramic	25-30



- To this point "composites" studied for nuclear systems have meant continuous fiber composites and have been limited to carbon fiber composites and silicon fiber composites.

Reinforced Fired Adobe Composite



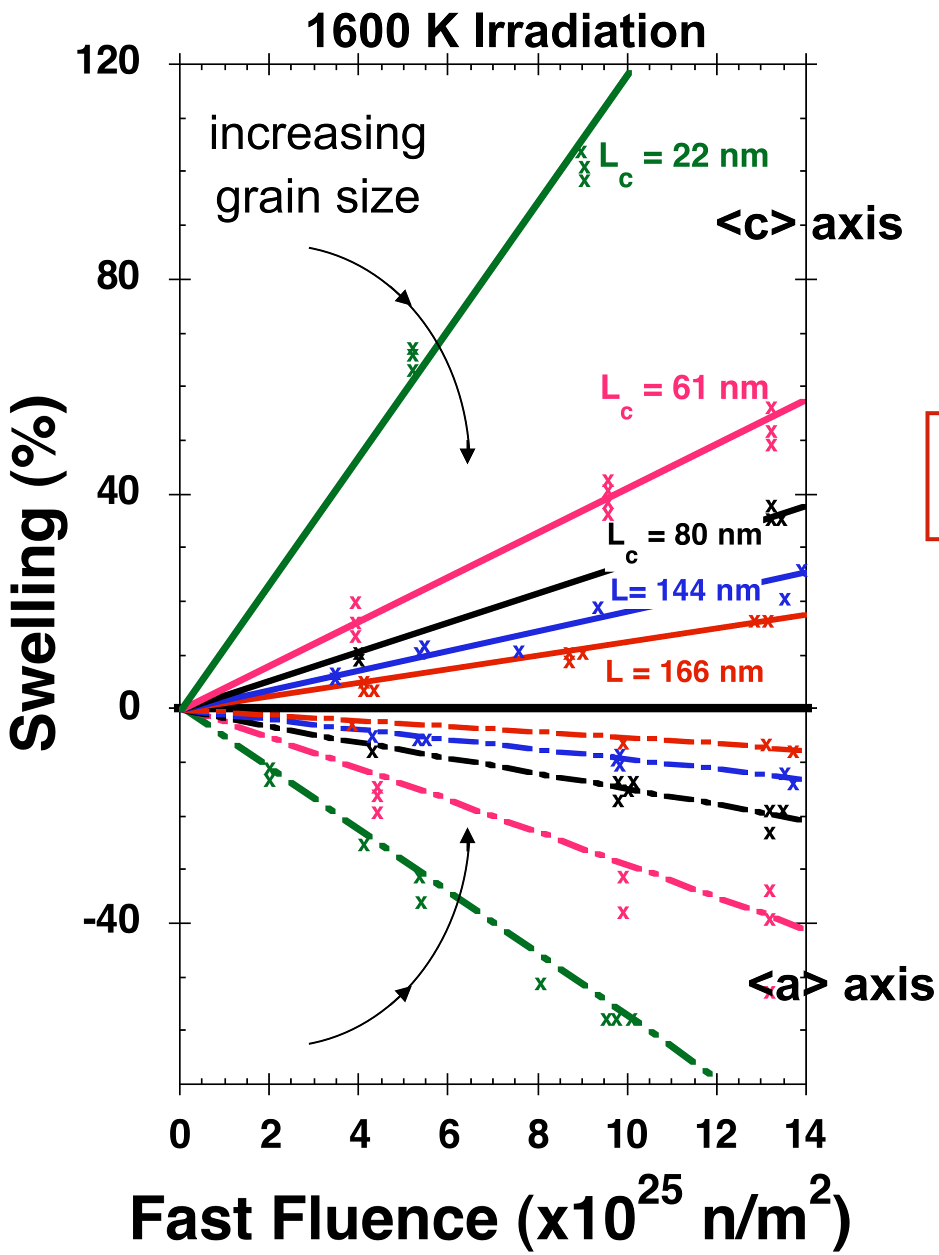
Puye Cliff Dwelling

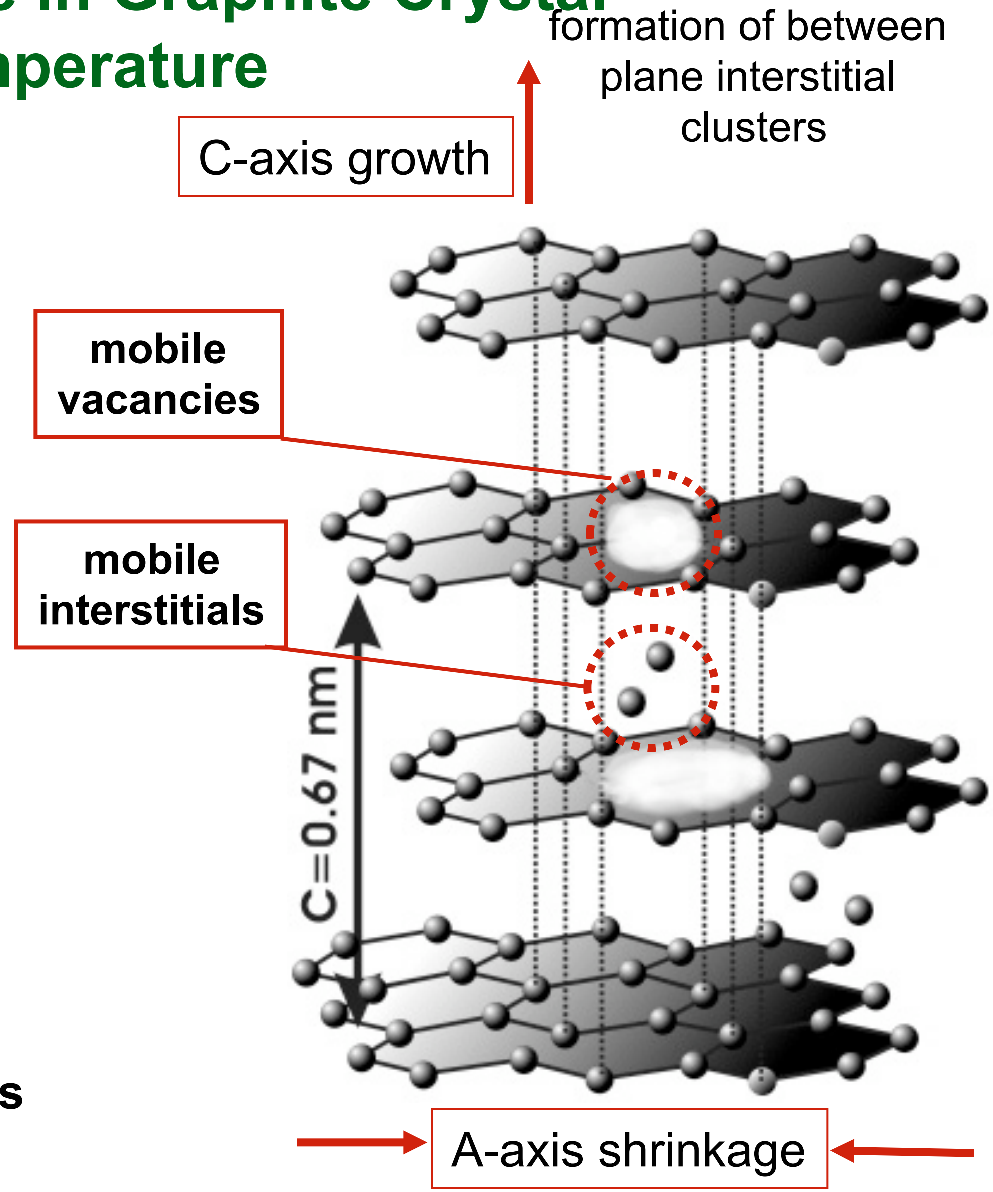
Anasaze Indians

1100-1580 AD

Tel-Dan Arch
~1600 BC

Dimensional Change in Graphite Crystal
High Temperature





- Overall Large Volume Increase
- Nearly Conserves Volume

carbon

1000K

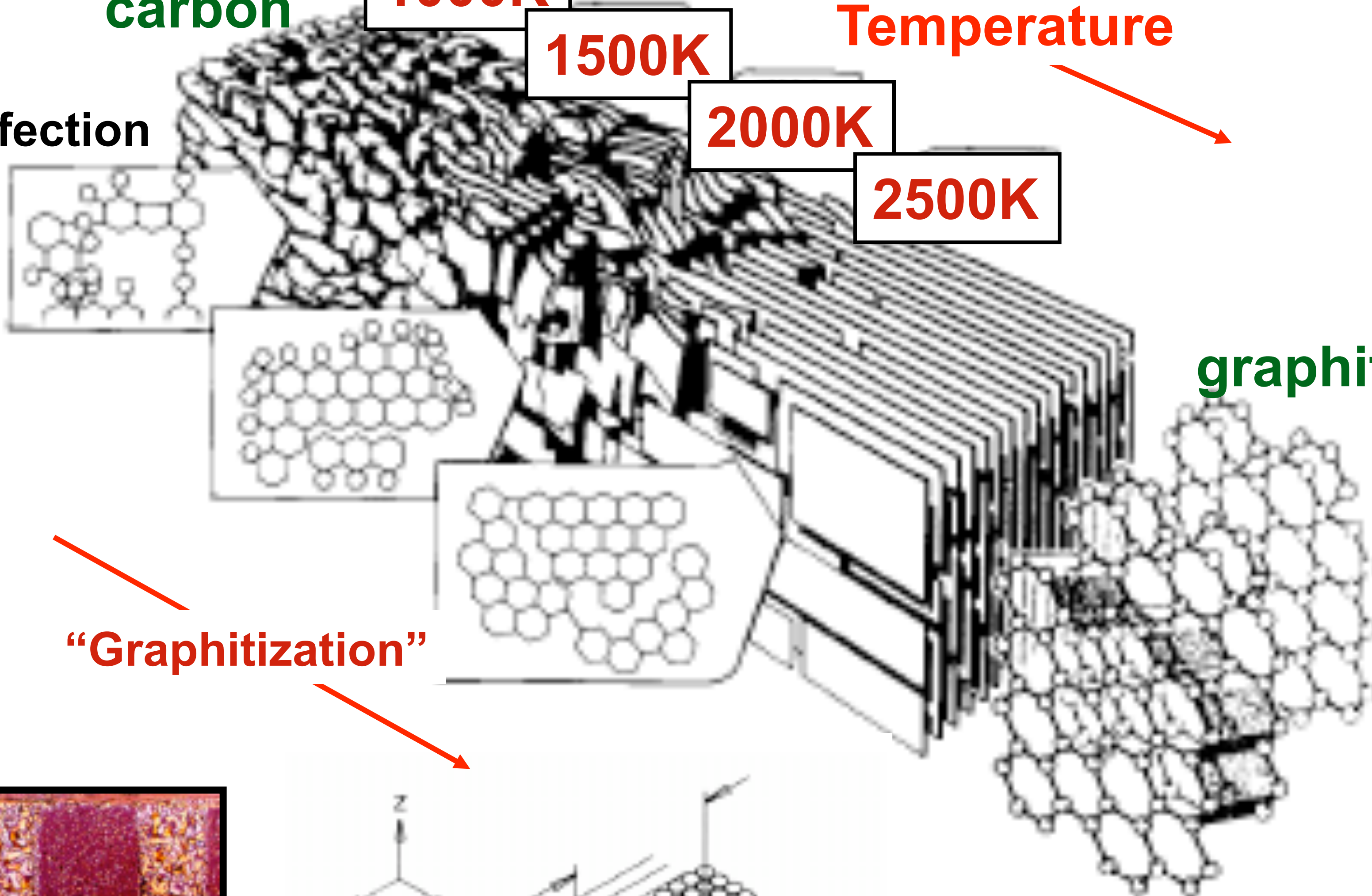
1500K

2000K

2500K

Temperature

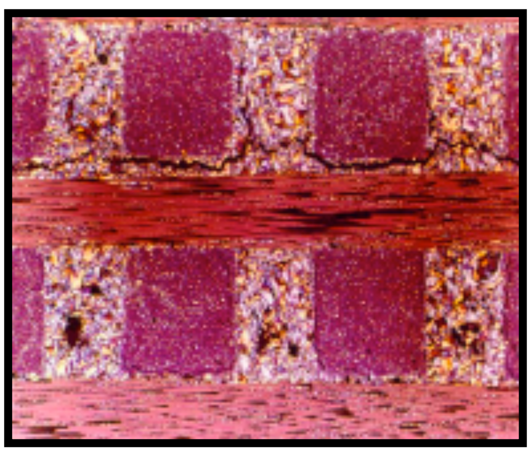
Low Perfection

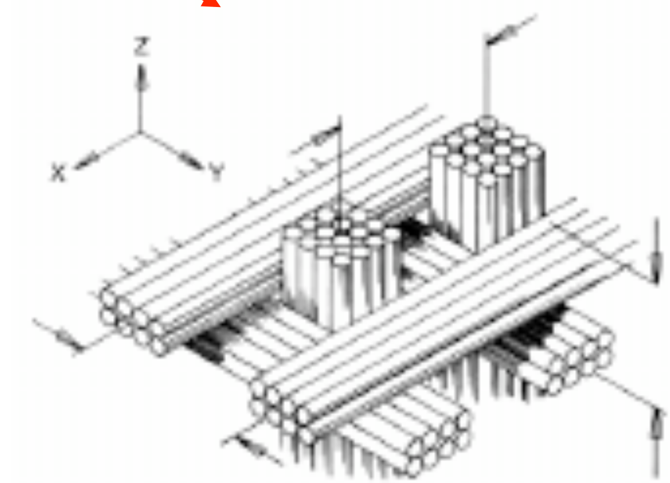


graphite

High Perfection

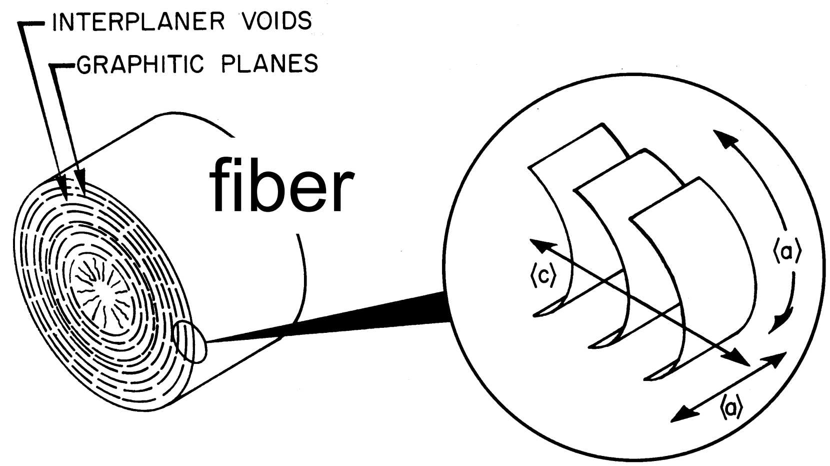
"Graphitization"



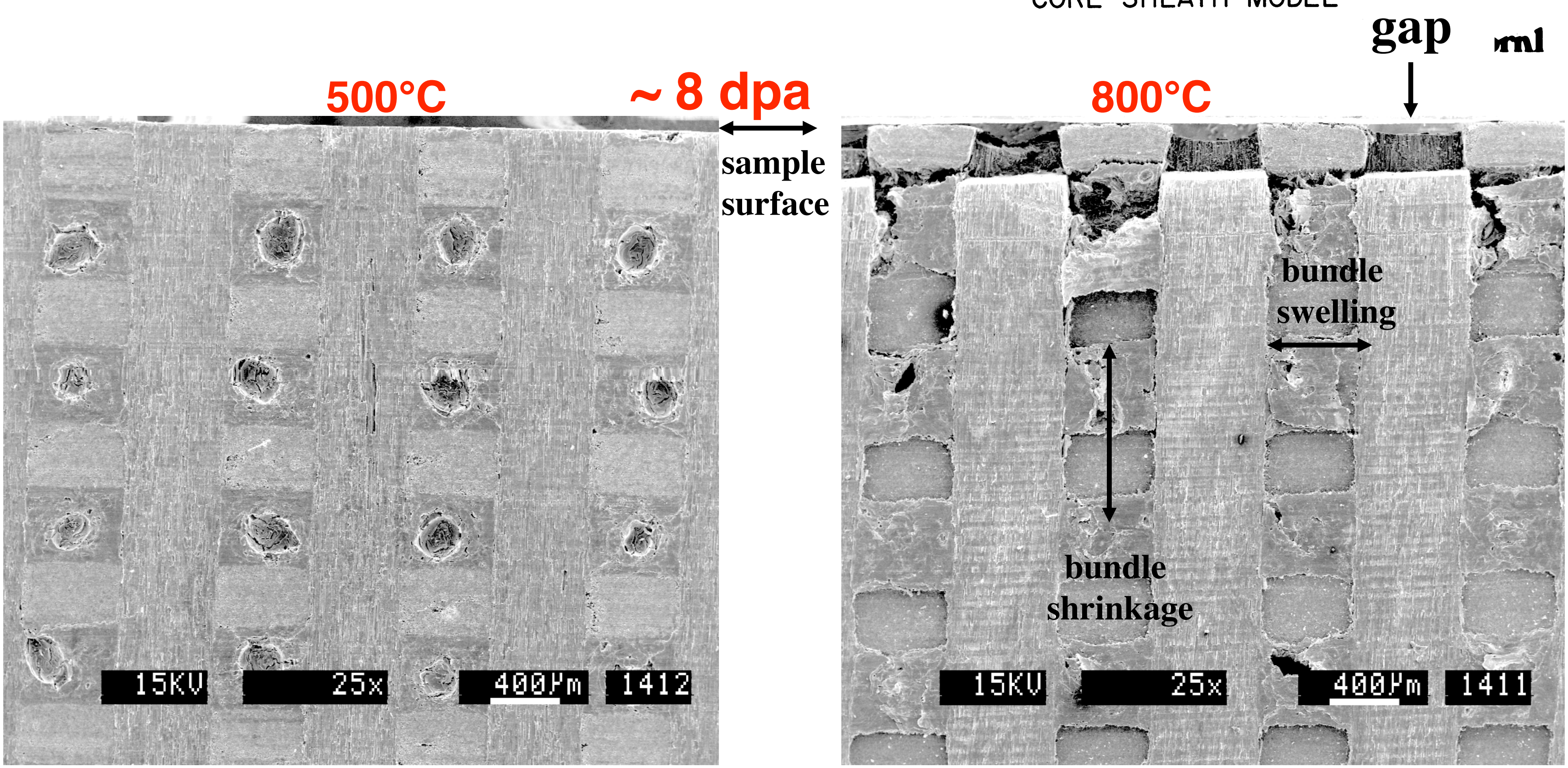


CFC's Under Irradiation

- composite irradiation-induced dimensional changes explained by classic graphite model…

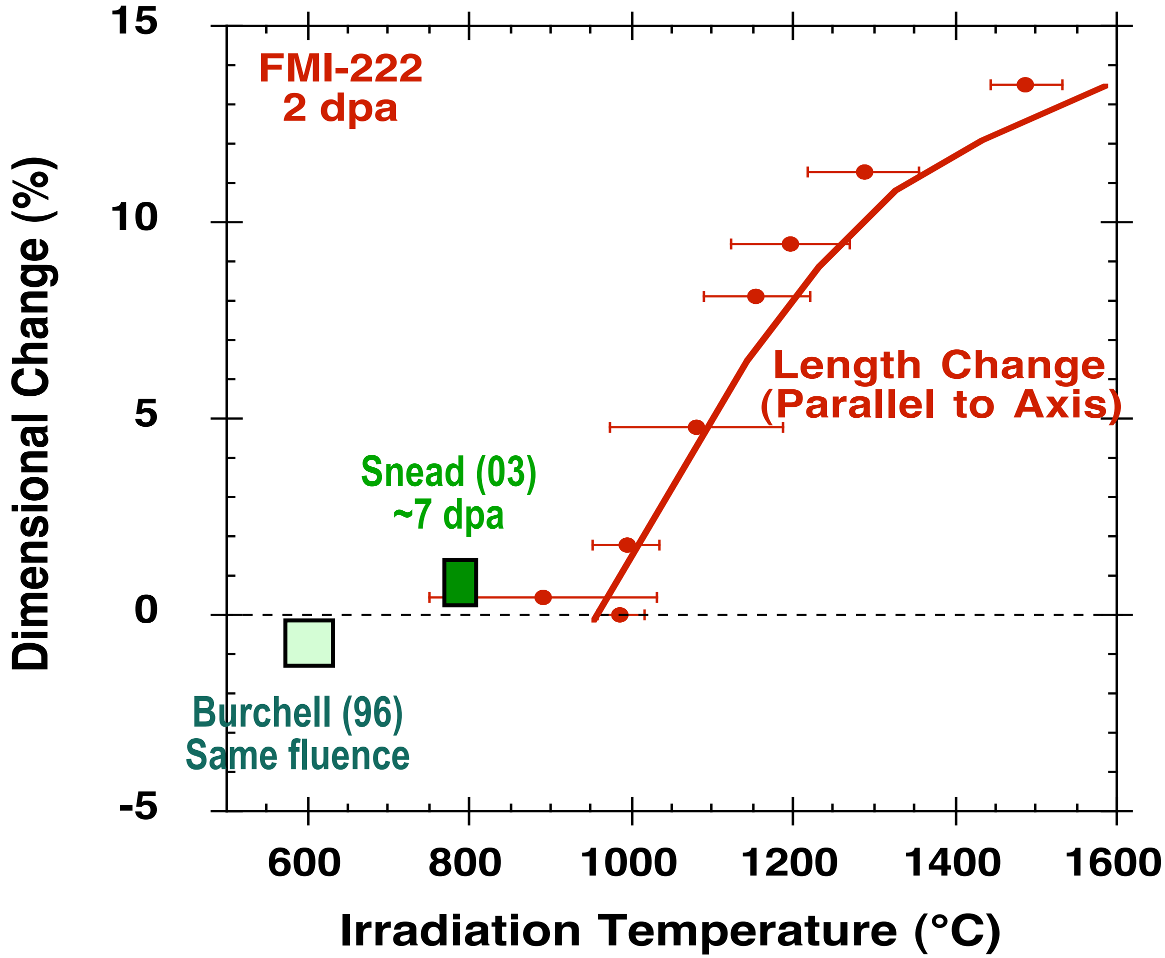


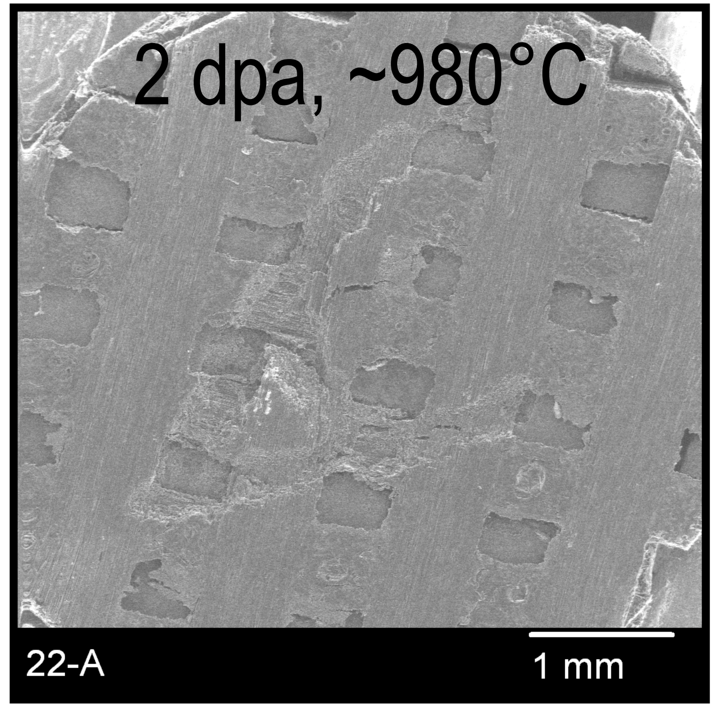
CORE-SHEATH MODEL

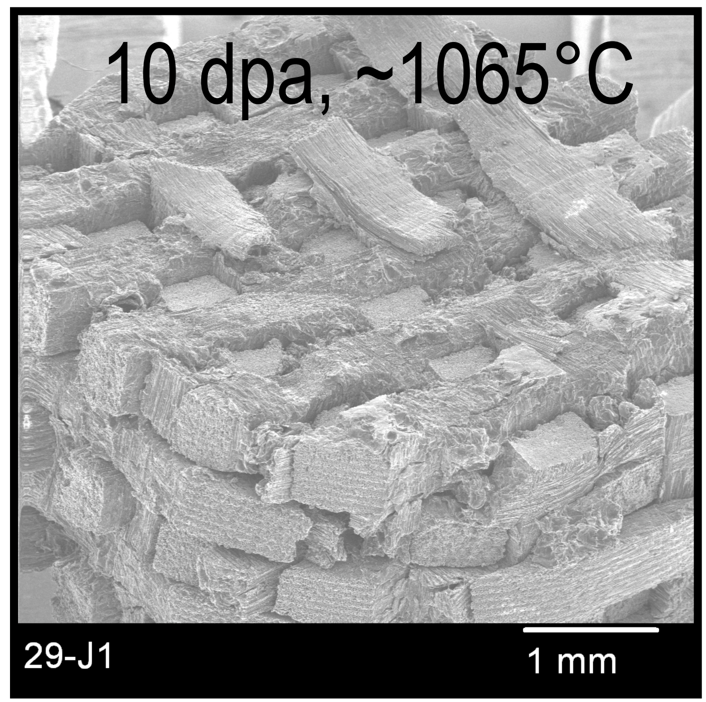


Dimensional Change in CFC

Elevated temperature irradiation begins to show severe dimensional instability

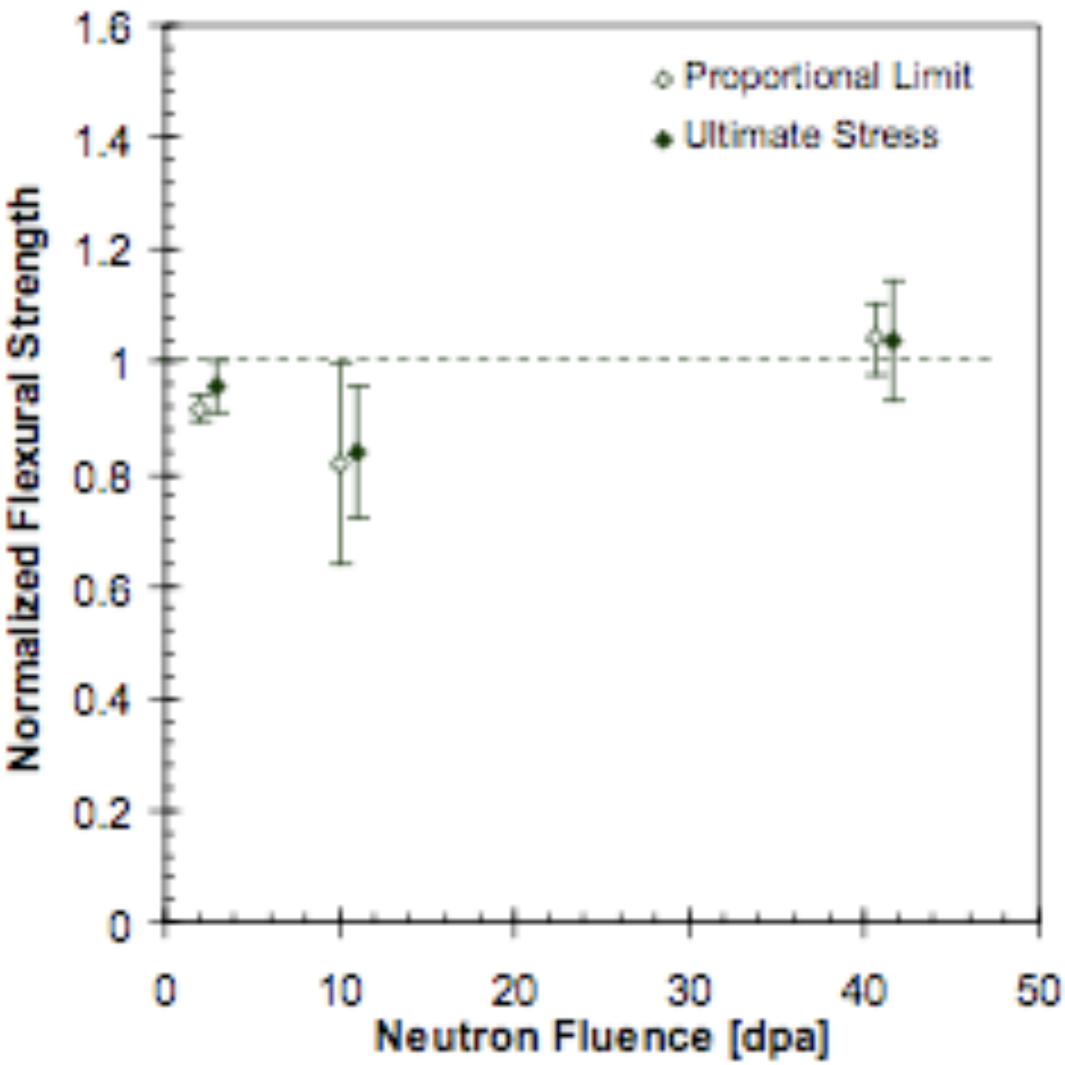






SiC/SiC Composites

- **SiC Composites have been under development for Fusion and Fission Reactor Systems for decades. The newest materials exhibit remarkable dimensional stability and retain physical properties to 42 dpa.**



Thermal conductivity is substantially lower than CFC

Summary Comments on Graphite and CFC's

- Graphite, like beryllium oxide, suffers from "crystal-driven" irradiation-induced changes. The effects can be somewhat mitigated, but strict lifetimes exist.

- CFC irradiation damage follows the same principals as for graphite.
  - Swelling perpendicular to basal planes, shrinkage within planes
  - Increase in strength and modulus up to composite lifetime.

- The lifetime of the composite will depend sensitively on the irradiation temperature and dose. It appears that, due to the inherent perfection of graphite fibers, the lifetime of the composite may be lower than that of nuclear graphite, especially at high temperatures.

- The CFC materials studied to date have been selected based on high intrinsic perfection (thermal conductivity.) This selection may lead to lower lifetime. --> we may do much better with poorer materials…

Comments on SiC Composite

• The most critical step in the development of SiC composites for nuclear applications was the development of an irradiation stable material. It now appears that we have a candidate material system for structural application, though prototypical irradiations (i.e., lifetime neutron dose, temperature, architecture, stress levels, etc) are just beginning.

• Now that we are moving from the proof-of-principal phase into the application phase for SiC/SiC a more comprehensive program has been initiated which includes:

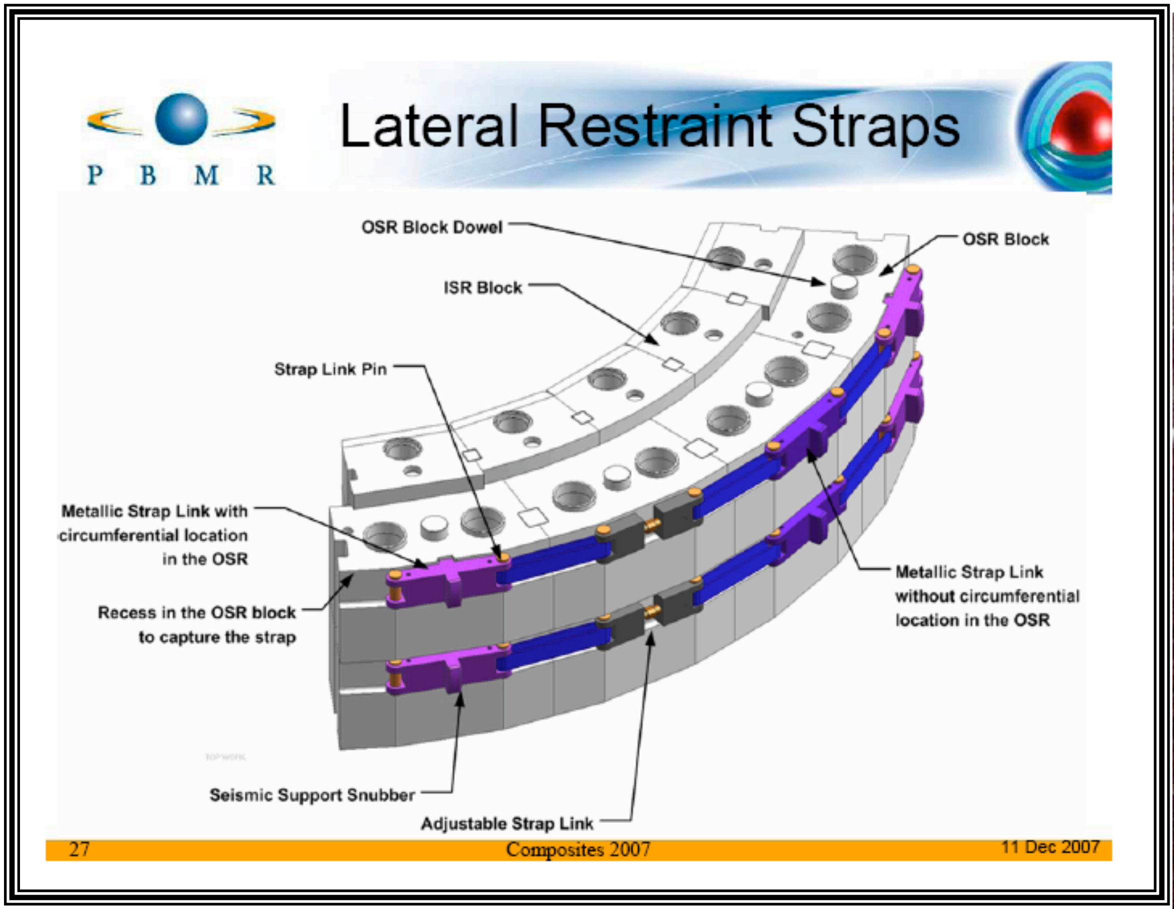
- Fundamental understanding of engineering properties such as swelling, thermal conductivity, irradiation creep, environmental corrosion, etc.
- Demonstration of scalability and component Q/A.
- Development of ASTM accepted testing standards and coordination with ASME for codification of the material.

• The ultimate choice between Carbon Fiber Composite and SiC/SiC will primarily depend on the dose and temperature of application:

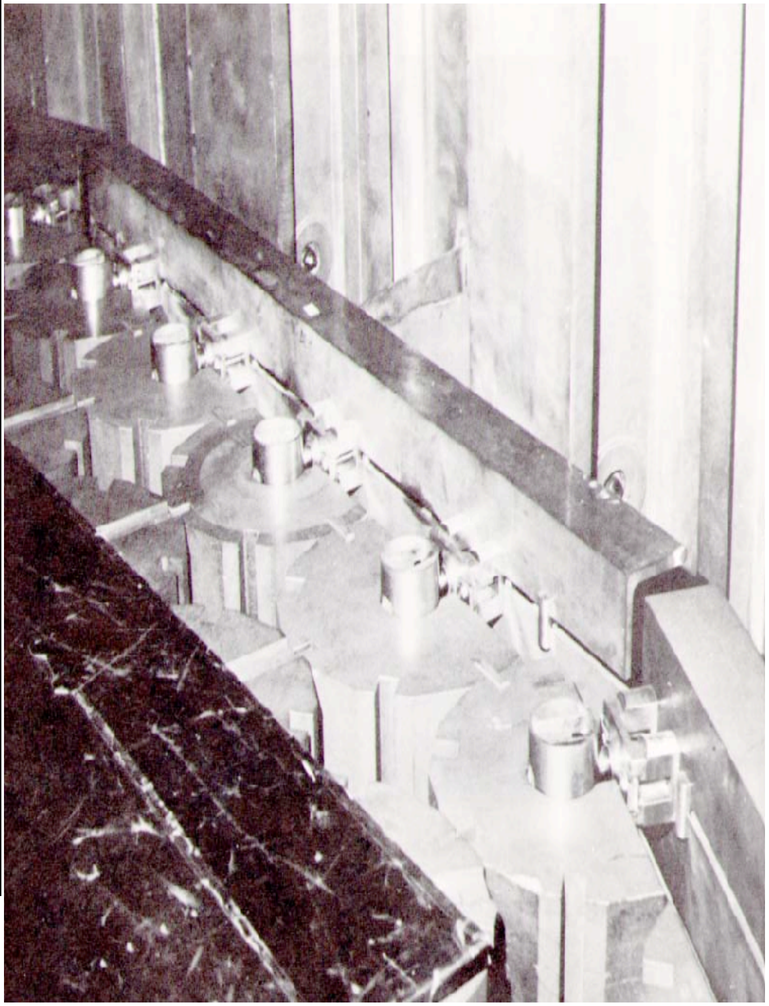
- CFC's : High Irradiated Thermal Conductivity (>50 W/m-K)
  Limited Lifetime (10 dpa for T , ~ 700°C) (~ 1 dpa T>1000°C)
- SiC/SiC : Poor thermal conductivity (< 5 W/m-K)
  Apparent Long Lifetime (> 20 dpa, T< 1300°C)

Core Restraint Systems AGR -v- HTGR

PBMR-**HTGR**
Carbon Fiber Composite and Steel

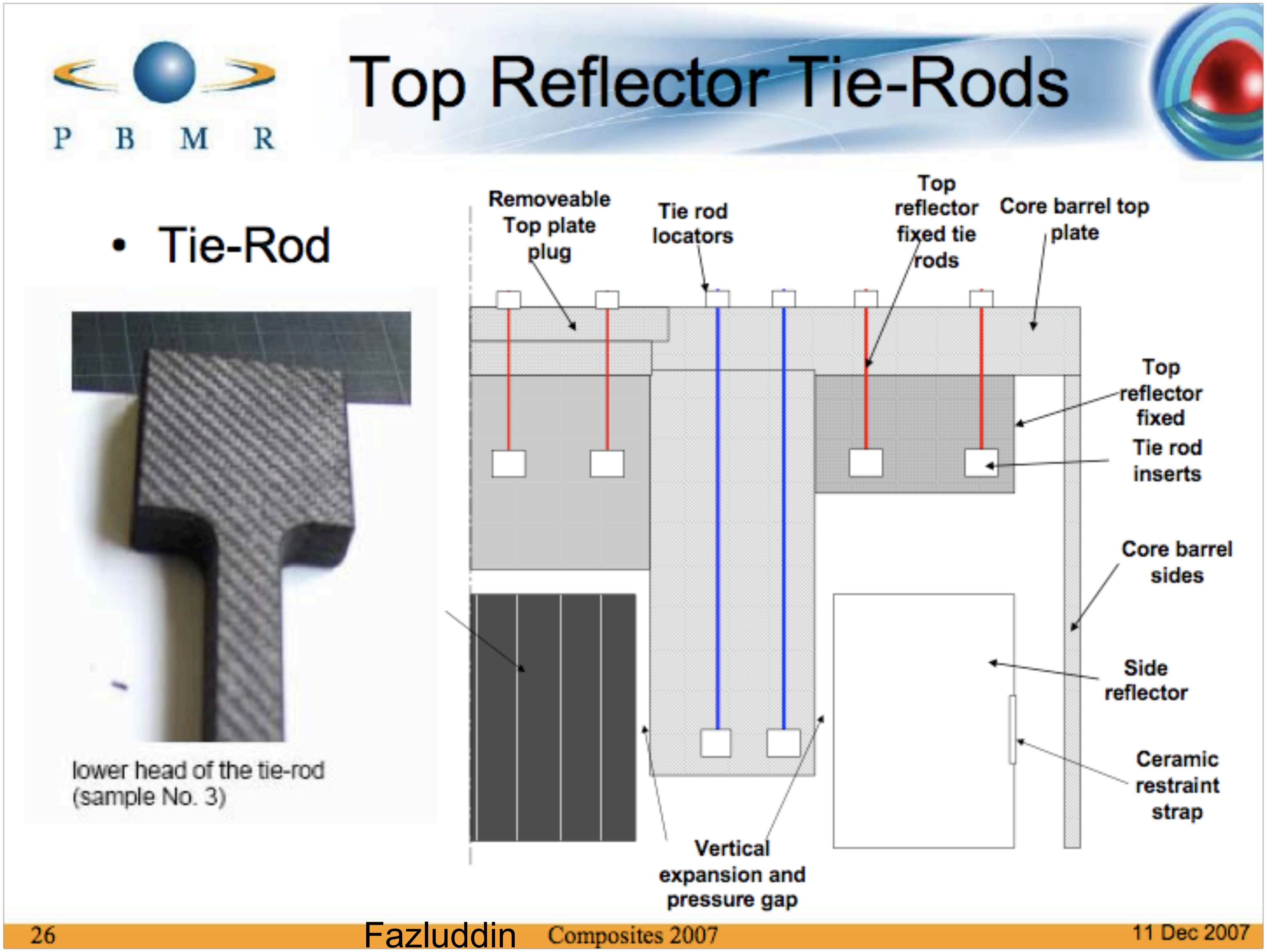


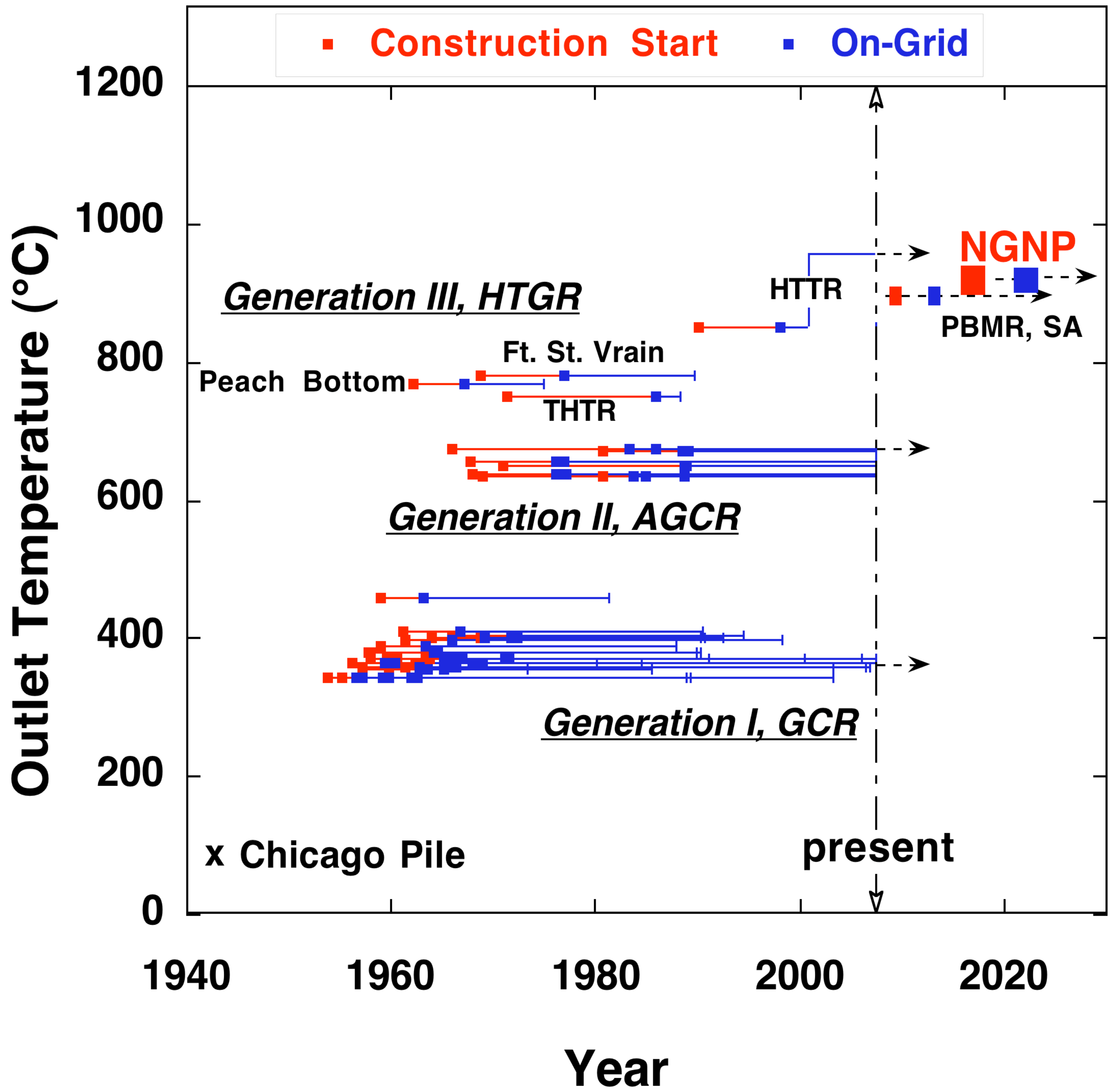
AGR
Steel Tank with Links



Lateral Restraint Strap

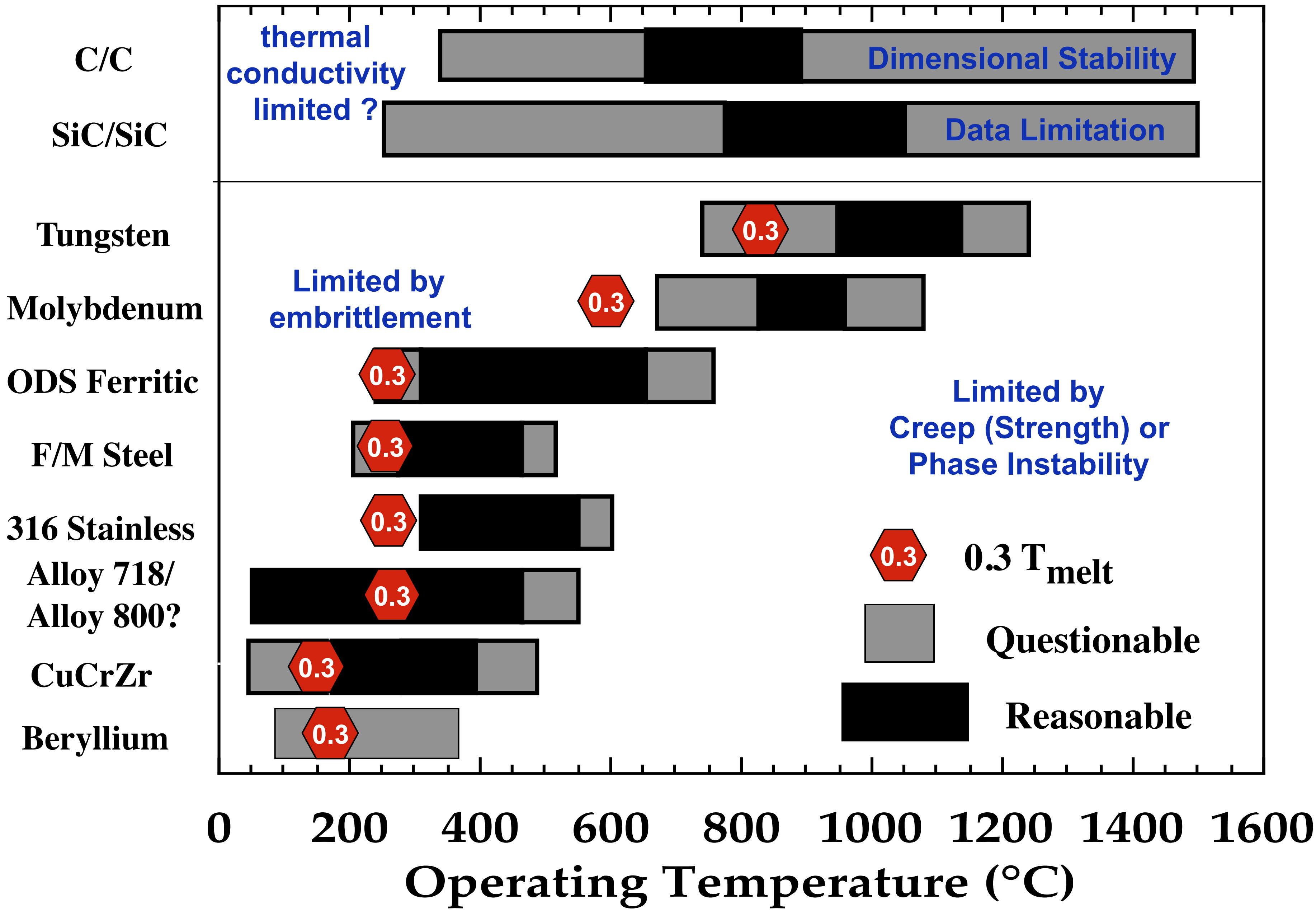
“Class 1 : Intermediate Temperature, Low-Dose” : PBMR Tie-rod

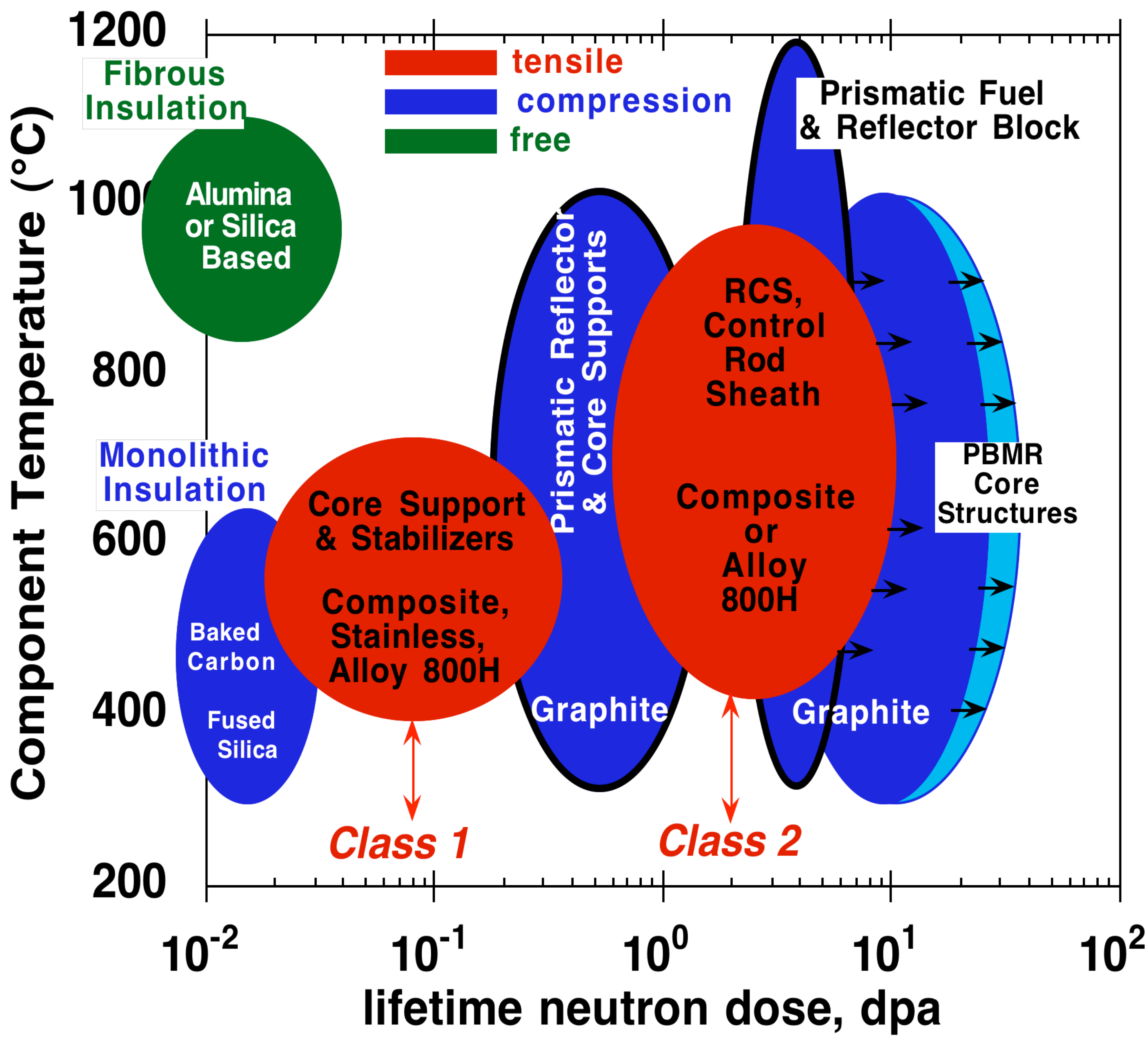




Construction Start
On-Grid
1200
1000
800
600
400
200
0
Outlet Temperature (°C)
Generation III, HTGR
HTTR
NGNP
PBMR, SA
Ft. St. Vrain
Peach Bottom
THTR
Generation II, AGCR
Generation I, GCR
x Chicago Pile
present
1940
1960
1980
2000
2020
Year

Operating Range, Irradiated Structural Materials





Two classes of in-core load bearing components:

Class 1 : intermediate temperature, low dose

Class 2 : intermediate to high temperature, intermediate dose


Questions ???